U.S. NUCLEAR REGULATORY COMMISSION

DIRECTIVE TRANSMITTAL

TN: DT-02-04

To:	NRC Management Directives Custodians
Subject:	Transmittal of Management Directive 5.12, "International Nuclear Event Scale Participation"
Purpose:	Directive and Handbook 5.12 are being issued to define NRC participation in the International Nuclear Event Scale (INES), as described in SECY-01-0071. MD 5.12 incorporates the guidance provided by the International Atomic Energy Agency (IAEA) regarding the use of the INES, as contained in "The INES User's Manual," which serves as Handbook 5.12.
Office and Division of Origin:	Incident Response Operations
Contact:	Robert Stransky, 415-6411
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Directive:	5.12 International Nuclear Event Scale Participation
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International Nuclear Event Scale Participation

Directive 5.12

Contents

International Nuclear Event Scale Participation
Directive 5.12

Policy
(5.12-01)

It is the policy of the U.S. Nuclear Regulatory Commission to participate in the International Nuclear Event Scale (INES), jointly developed by the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (NEA/OECD). This participation includes the evaluation of all events (reactor, fuel facility, and licensed materials), except medical misadministrations, reported to the NRC Operations Center for potential reporting using the INES. Reports are issued for all events rated as level 2 or higher, or as requested by another INES member nation.

Objective
(5.12-02)

To ensure that the rating of events involving reactor and fuel cycle facilities and NRC or Agreement State licensed materials using the INES is performed in accordance with the guidance provided jointly by the IAEA and NEA/OECD.

Organizational Responsibilities and Delegations of Authority
(5.12-03)

Executive Director for Operations (EDO)
(031)

- Oversees NRC participation in the International Nuclear Event Scale. (a)

- Ensures the cooperation and support of all NRC offices in providing cognizant personnel to support the rating of events using the INES. (b)

Director, Incident Response Operations (IRO)
(032)

- Ensures that INES ratings are performed accurately, efficiently, and in a timely manner. (a)

- Designates an “INES National Officer” to represent NRC in matters involving the further development and use of the INES. (b)

- Calls on other NRC offices to provide resources needed to quickly and accurately rate events using the INES. (c)

- Provides IRO personnel to coordinate the rating of events using the INES and the dissemination of these ratings to the IAEA and other member nations. (d)

- Supports training for individuals performing INES rating of events. (e)

Organizational Responsibilities and Delegations of Authority
(5.12-03) (continued)

Director, Incident Response Operations (IRO)
(032) (continued)

- Coordinates with other Federal Government agencies through various interagency working groups in matters involving the use of the INES. (f)

Director, Office of Nuclear Material Safety and Safeguards (NMSS)
(033)

- Provides personnel to rate events involving fuel cycle facilities and licensed radioactive material using the INES. (a)

- Supports training for individuals performing INES rating of events. (b)

- Provides office procedures for the rating of events involving fuel cycle facilities and licensed radioactive material, except for medical misadministrations. (c)

Director, Office of Nuclear Reactor Regulation (NRR)
(034)

- Provides personnel to rate events involving nuclear reactor facilities using the INES. (a)

- Supports training for individuals performing INES rating of events. (b)

Organizational Responsibilities and Delegations of Authority
(5.12-03) (continued)

Director, Office of Nuclear Reactor Regulation (NRR)
(034) (continued)

- Provides office procedures for the rating of events involving nuclear reactor facilities. (c)

Director, Office of State and Tribal Programs (STP)
(035)

Coordinates with NMSS and Agreement States to provide for timely and accurate rating of events occurring in Agreement States.

Director, Office of Public Affairs (OPA)
(036)

Responds to requests from the news media and the general public about the INES in general and regarding the rating of specific events, normally within 2 working days after the event.

Definition
(5.12-04)

International Nuclear Event Scale (INES). The INES is a means for communicating the safety significance of events at nuclear facilities to the public in consistent terms, using a numerical scale that ranges from 0 (no safety significance) to 7 (major accident).

Applicability
(5.12-05)

The policy and guidance in this directive apply to all NRC employees.

Implementation Guidelines
(5.12-06)

In general, the NRC's implementation of the INES will be performed in accordance with the guidance and rating methodology provided in "The International Nuclear Event Scale (INES) User's Manual" (Handbook 5.12). However, several additional points regarding the NRC's participation are summarized below:

- Review of events for potential reporting using the INES will be incorporated into the normal event assessment process of the respective program office. (a)

- The methodology for reviewing events for possible reporting using the INES will generally follow the methodology provided in the INES User's Manual. However, each program office will be responsible for developing specific review criteria. Slight deviations from the specific guidance included in the INES User's Manual is acceptable if approved by the director of the program office. (b)

- Reports will be submitted only for those events rated at level 2 or higher on the INES, unless specifically requested by the IAEA or another INES member nation. (c)

- Draft reports require approval by a first-tier Senior Executive Service manager (i.e., a branch chief). (d)

- The INES National Officer, or designee, shall submit INES reports to the IAEA. (e)

Handbook

(5.12-07)

"The International Nuclear Event Scale (INES) User's Manual," published jointly by the IAEA and NEA/OECD, is the handbook for this directive. It contains information regarding the rating of events using the INES.

References

(5.12-08)

SECY-01-0071, "Expanded NRC Participation in the Use of the International Nuclear Event Scale."

Staff Requirements Memorandum/SECY-01-0071, "Expanded NRC Participation in the Use of the International Nuclear Event Scale."

"The International Nuclear Event Scale (INES) User's Manual," IAEA-INES-2001, 2001 Edition or more current version, if available.

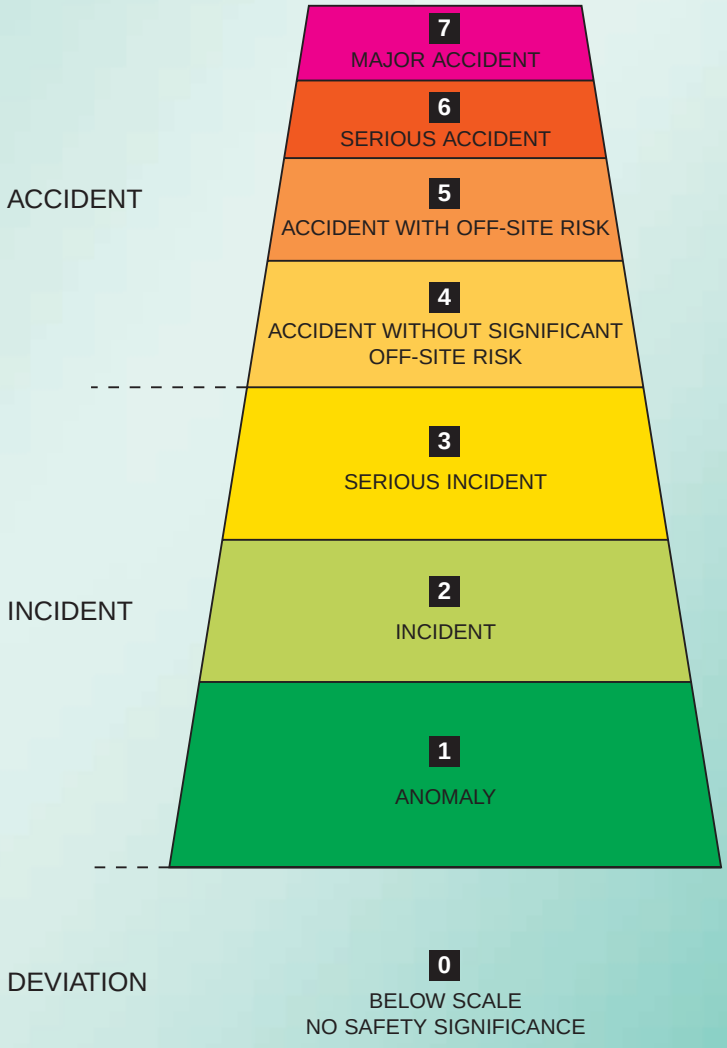
International Nuclear Event Scale Participation

Handbook 5.12

The International Nuclear Event Scale (INES)

User's Manual

2001 Edition



JOINTLY PREPARED BY IAEA AND OECD/NEA

THE INTERNATIONAL NUCLEAR EVENT SCALE (INES) USER'S MANUAL

2001 EDITION

Jointly prepared by IAEA and OECD/NEA

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INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2001

THE INTERNATIONAL NUCLEAR EVENT SCALE (INES)
USER'S MANUAL
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February 2001

FOREWORD

The International Nuclear Event Scale (INES) was introduced in March 1990 jointly by the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD/NEA). Its primary purpose is to facilitate communication and understanding between the nuclear community, the media and the public on the safety significance of events occurring at nuclear installations. The scale was refined in 1992 in the light of experience gained and extended to be applicable to any event associated with radioactive material and/or radiation, including the transport of radioactive materials.

This edition of the INES User's Manual incorporates experience gained from applying the 1992 version of the scale and the document entitled "Clarification of Issues Raised". As such, it replaces those earlier publications. It does not amend the technical basis of the INES rating procedure but is expected to facilitate the task of those who are required to rate the safety significance of events using the INES scale.

The INES communication network currently receives and disseminates event information to the INES National Officers of 60 Member States on special Event Rating Forms which represent official information on the events, including the rating. The INES communication process has led each participating country to set up an internal network which ensures that all events are promptly communicated and rated whenever they have to be reported outside or inside the country.

The IAEA provides training services on the use of INES on request.

CONTENTS

Part I

SUMMARY DESCRIPTION

I–1. INTRODUCTION

I–1.1. Background

The International Nuclear Event Scale (INES) is a means for promptly communicating to the public in consistent terms the safety significance of events reported at nuclear installations. By putting events into proper perspective, it can facilitate common understanding among the nuclear community, the media and the public.

The scale was designed by an international group of experts convened jointly in 1989 by the IAEA and the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD/NEA). It also reflects the experience gained from the use of similar scales in France and Japan as well as from consideration of possible scales in several other countries.

Initially the scale was applied for a trial period to classify events at nuclear power plants, and then extended and adapted to enable it to be applied to all installations associated with the civil nuclear industry. It is now operating successfully in over 60 countries. This edition of the INES User's Manual can be applied to any event associated with radioactive material and/or radiation and to any event occurring during the transport of radioactive material.

I–1.2. General description of the scale

Events are classified on the scale at seven levels: the upper levels (4–7) are termed "accidents" and the lower levels (1–3) "incidents". Events which have no safety significance are classified below scale at level 0 and are termed "deviations". Events which have no safety relevance are termed "out of scale". The structure of the scale is shown in Fig. 1, in the form of a matrix with key words. The words used are not intended to be precise or definitive. Each level is defined in detail in Parts III and IV of this manual. Events are considered in terms of three different areas of impact represented by each of the columns: off-site impact, on-site impact and impact on defence in depth.

The first column relates to events resulting in off-site releases of radioactivity. Since this is the only possible direct impact on the public, such releases are understandably of particular concern. Thus, the lowest point in this column represents a release giving the critical group an estimated radiation dose numerically equivalent to about one-tenth of the annual dose limit for the public; this is classified as level 3.

	AREA OF IMPACT		
	OFF-SITE IMPACT	ON-SITE IMPACT	IMPACT ON DEFENCE IN DEPTH
7 MAJOR ACCIDENT	MAJOR RELEASE: WIDESPREAD HEALTH AND ENVIRONMENTAL EFFECTS		
6 SERIOUS ACCIDENT	SIGNIFICANT RELEASE: LIKELY TO REQUIRE FULL IMPLEMENTATION OF PLANNED COUNTERMEASURES		
5 ACCIDENT WITH OFF-SITE RISK	LIMITED RELEASE: LIKELY TO REQUIRE PARTIAL IMPLEMENTATION OF PLANNED COUNTERMEASURES	SEVERE DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS	
4 ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK	MINOR RELEASE: PUBLIC EXPOSURE OF THE ORDER OF PRESCRIBED LIMITS	SIGNIFICANT DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS/FATAL EXPOSURE OF A WORKER	
3 SERIOUS INCIDENT	VERY SMALL RELEASE: PUBLIC EXPOSURE AT A FRACTION OF PRESCRIBED LIMITS	SEVERE SPREAD OF CONTAMINATION/ACUTE HEALTH EFFECTS TO A WORKER	NEAR ACCIDENT NO SAFETY LAYERS REMAINING
2 INCIDENT		SIGNIFICANT SPREAD OF CONTAMINATION/ OVEREXPOSURE OF A WORKER	INCIDENTS WITH SIGNIFICANT FAILURES IN SAFETY PROVISIONS
1 ANOMALY			ANOMALY BEYOND THE AUTHORIZED OPERATING REGIME
0 DEVIATION	**NO SAFETY SIGNIFICANCE**		

FIG. 1. Basic structure of the scale (the criteria given in the matrix are broad indicators only).

Such a dose is also typically about one-tenth of the average annual dose received from natural background radiation. The highest level is a major nuclear accident with widespread health and environmental consequences.

The second column considers the on-site impact of the event. This category covers a range from level 2 (contamination and/or overexposure of a worker) to level 5 (severe damage to the reactor core or radiological barriers).

All nuclear facilities are designed and operated so that a succession of safety layers act to prevent major off-site or on-site impact and the extent of the safety layers provided generally will be commensurate with the potential for such impacts. These safety layers must all fail before substantial off-site or on-site consequences occur. The provision of these layers is termed “defence in depth”. The third column relates to incidents in which these defence in depth provisions have been degraded. This column spans the incident levels from 1 to 3.

An event which has an impact on more than one area is always rated at the highest level identified. Events which do not reach the threshold in any of the three areas are rated below scale at level 0. Figure 2 gives typical descriptions of events at each level together with examples of the rating of nuclear events which have occurred in the past at nuclear installations.

I–1.3. Scope of the scale

The scale can be applied to any event associated with radioactive material and/or radiation and to any event occurring during the transport of radioactive material. It does not classify industrial accidents or other events which are not related to nuclear or radiological operations. Such events are termed “out of scale”. For example, although events associated with a turbine or generator can affect safety related equipment, faults affecting only the availability of a turbine or generator would be classified as out of scale. Similarly, events such as fires would be classified as out of scale if they did not involve any possible radiological hazard and did not affect the safety layers.

The scale does not apply to those controls provided only for the safeguarding of fissile material. Equally, published accountancy imbalances for fissile material (material unaccounted for (MUF)) would be classified as out of scale.

I–1.4. Using the scale

Although broadly comparable, nuclear and radiological safety criteria and the terminology used to describe them vary from country to country. The international scale has been designed to take account of this fact, but it is possible that user countries may wish to clarify the scale within their national context.

The detailed rating procedures are provided in this manual. The INES leaflet should not be used as the basis for rating events as it only provides examples of events at each level, rather than actual definitions.

The scale is designed for prompt use following an event. However, there will be occasions when a longer time-scale is required to understand and rate the consequences of an event. In these rare circumstances, a provisional rating will be given with confirmation at a later date. It is also possible that as a result of further information, an event may require re-rating.

Although the scale is used for all facilities, it is physically impossible at some types of installation for events to occur which involve the release to the environment of considerable quantities of radioactive material. For these installations, the upper levels of the scale would not be applicable. These include research reactors, unirradiated nuclear fuel treatment facilities and waste storage sites.

LEVEL/ DESCRIPTOR	NATURE OF THE EVENTS	EXAMPLES
7 MAJOR ACCIDENT	• External release of a large fraction of the radioactive material in a large facility (e.g. the core of a power reactor). This would typically involve mixture of short and long lived radioactive fission products (in quantities radiologically equivalent to more than tens of thousands of terabecquerels of ^{131}I). Such a release would result in the possibility of acute health effects; delayed health effects over a wide area, possibly involving more than one country; long term environmental consequences.	Chernobyl nuclear power plant, USSR (now in Ukraine), 1986
6 SERIOUS ACCIDENT	• External release of radioactive material (in quantities radiologically equivalent to the order of thousands to tens of thousands of terabecquerels of ^{131}I). Such a release would be likely. to result in full implementation of countermeasures covered by local emergency plans to limit serious health effects.	Kyshtym Reprocessing Plant, USSR (now in Russian Federation), 1957
5 ACCIDENT WITH OFF-SITE RISK	• External release of radioactive material (in quantities radiologically equivalent to the order of hundreds to thousands of terabecquerels of ^{131}I). Such a release would be likely to result in partial implementation of countermeasures covered by emergency plans to lessen the likelihood of health effects. • Severe damage to the installation. This may involve severe damage to a large fraction of the core of a power reactor, a major criticality accident or a major fire or explosion releasing large quantities of radioactivity within the installation.	Windscale Pile, UK, 1957 Three Mile Island nuclear power plant, USA, 1979
4 ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK	• External release of radioactivity resulting in a dose to the critical group of the order of a few millisieverts.[a] With such a release the need for off-site protective actions would be generally unlikely except possibly for local food control. • Significant damage to the installation. Such an accident might include damage leading to major on-site recovery problems such as partial core melt in a power reactor and comparable events at non-reactor installations. • Irradiation of one or more workers resulting in an overexposure where a high probability of early death occurs.	Windscale Reprocessing Plant, UK, 1973 Saint Laurent nuclear power plant, France, 1980 Buenos Aires Critical Assembly, Argentina, 1983
3 SERIOUS INCIDENT	• External release of radioactivity resulting in a dose to the critical group of the order of tenths of millisieverts.[a] With such a release, off-site protective measures may not be needed. • On-site events resulting in doses to workers sufficient to cause acute health effects and/or an event resulting in a severe spread of contamination for example a few thousand terabecquerels of activity released in a secondary containment where the material can be returned to a satisfactory storage area. • Incidents in which a further failure of safety systems could lead to accident conditions, or a situation in which safety systems would be unable to prevent an accident if certain initiators were to occur.	Vandellos nuclear power plant, Spain, 1989
2 INCIDENT	• Incidents with significant failure in safety provisions but with sufficient defence in depth remaining to cope with additional failures. These include events where the actual failures would be rated at level 1, but which reveal significant additional organizational inadequacies or safety culture deficiencies. • An event resulting in a dose to a worker exceeding a statutory annual dose limit and/or an event which leads to the presence of significant quantities of radioactivity in the installation in areas not expected by design and which require corrective action.	
1 ANOMALY	• Anomaly beyond the authorized regime, but with significant defence in depth remaining. This may be due to equipment failure, human error or procedural inadequacies and may occur in any area covered by the scale, e.g. plant operation, transport of radioactive material, fuel handling, and waste storage. Examples include: breaches of technical specifications or transport regulations, incidents without direct safety consequences that reveal inadequacies in the organizational system or safety culture, minor defects in pipework beyond the expectations of the surveillance programme.	
DEVIATION 0	• Deviations where operational limits and conditions are not exceeded and which are properly managed in accordance with adequate procedures. Examples include: a single random failure in a redundant system discovered during periodic inspections or tests, a planned reactor trip proceeding normally, spurious initiation of protection systems without significant consequences, leakages within the operational limits, minor spreads of contamination within controlled areas without wider implications for safety culture..	

[a] The doses are expressed in terms of effective dose equivalent (whole dose body). Those criteria, where appropriate, can also be expressed in terms of corresponding annual effluent discharge limits authorized by national authorities.

FIG. 2. The International Nuclear Event Scale (for prompt communication of safety significance).

The scale does not replace the criteria already adopted nationally and internationally for the technical analysis and reporting of events to safety authorities. Nor does it form a part of the formal emergency arrangements that exist in each country to deal with radiological accidents.

The scale is not appropriate as the basis for selecting events for feedback of operational experience, as important lessons can often be learnt from events of relatively minor significance.

Finally, it is not appropriate to use this scale to compare safety performance between countries. Each country has different arrangements for reporting minor events to the public, and it is difficult to ensure precise international consistency in rating events at the boundary between level 0 and level 1. Although information will be available generally on events at level 2 and above on the scale, the statistically small number of such events, which also varies from year to year, makes it difficult to provide meaningful international comparisons.

I–1.5. Examples of rated nuclear events

The 1986 accident at the Chernobyl nuclear power plant in the USSR (now in Ukraine) had widespread environmental and human health effects. It is rated at level 7.

The 1957 accident at the Kyshtym reprocessing plant in the USSR (now in the Russian Federation) led to a large off-site release. Emergency measures, including evacuation of the population, were taken to limit serious health effects. On the basis of the off-site impact, this event is rated at level 6.

The 1957 accident at the air cooled graphite reactor pile at the Windscale (now Sellafield) facility in the United Kingdom involved an external release of radioactive fission products. On the basis of the off-site impact, it is rated at level 5.

The 1979 accident at the Three Mile Island nuclear power plant in the USA resulted in a severely damaged reactor core. The off-site release of radioactivity was very limited. The event is rated at level 5 on the basis of the on-site impact.

The 1973 accident at the Windscale (now Sellafield) reprocessing plant in the United Kingdom involved a release of radioactive material into a plant operating area as a result of an exothermic reaction in a process vessel. It is rated at level 4 on the basis of the on-site impact.

The 1980 accident at the Saint Laurent nuclear power plant in France resulted in partial damage to the reactor core, but there was no external release of radioactivity. It is rated at level 4 on the basis of the on-site impact.

The 1983 accident at the RA-2 critical assembly in Buenos Aires, Argentina, an accidental power excursion owing to non-observance of safety rules during a core modification sequence, resulted in the death of the operator, who was probably

3–4 m away. Assessments of the doses absorbed indicate 21 Gy for the gamma dose, together with 22 Gy for the neutron dose. The event is rated at level 4 on the basis of the on-site impact.

The 1989 incident at the Vandellos nuclear power plant in Spain did not result in an external release of radioactivity, nor was there damage to the reactor core or contamination on site. However, the damage to the plant's safety systems from the fire degraded the defence in depth significantly. The event is rated at level 3 on the basis of the impact on defence in depth.

The vast majority of reported events are rated below level 3. Although no examples of these events are given here, countries using the scale may individually wish to provide examples of events at these lower levels.

I–1.6. Structure of the manual

This manual consists of six parts:

— Part I provides an overview of the scale,
— Part II is a summary of the procedure to be used to rate events and to report them to the INES information service,
— Part III gives the detailed guidance required to rate events in terms of off-site and on-site impact,
— Part IV provides the detailed guidance required to rate events in terms of their impact on defence in depth,
— Part V consists of examples to illustrate the use of the rating guidance,
— Part VI contains a number of appendices giving detailed information on particular aspects of the scale.

Part II

RATING PROCEDURE AND REPORTING EVENTS TO THE IAEA

II–1. RATING PROCEDURE

The flow chart provided on the following pages briefly describes the INES rating procedure for rating any event associated with radioactive material and/or radiation and any event occurring during the transport of radioactive material. The format of the flow chart is intended to show the logical route to be followed to assess the safety significance of any event. It provides an overview for those new to rating events and a summary of the procedure for those familiar with the INES User's Manual. It cannot, of course, be used in isolation from the detailed guidance provided in Parts III and IV. The computer software INESAR (INES Automatic Rating) has been developed on the basis of a similar earlier flow chart.

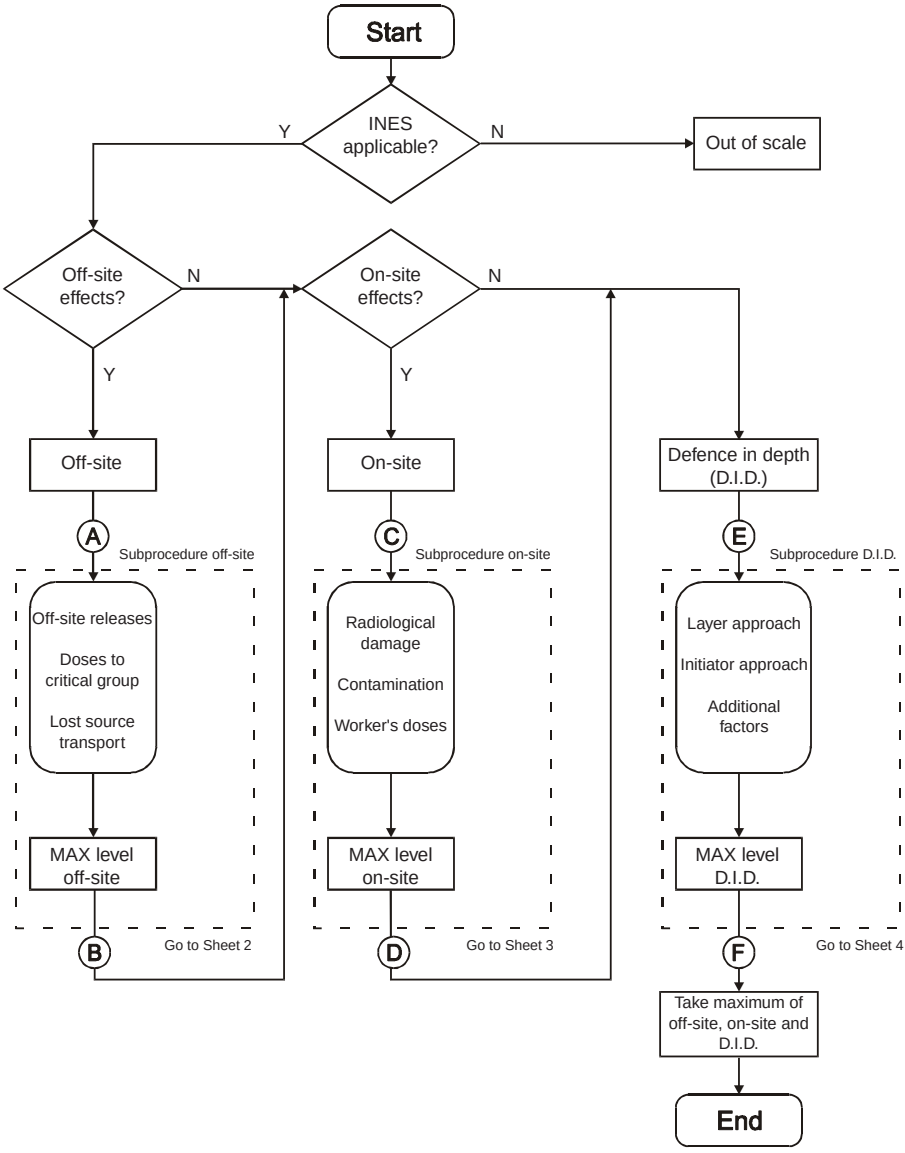
II–2. COMMUNICATING EVENTS TO THE IAEA INFORMATION SERVICE

The INES National Officer is committed to communicate as quickly as possible (target: within 24 hours) official information on the consequences of an event to all the participating countries (see Appendix VI) through the IAEA INES Information Service. The criteria for identifying which events should be communicated are:

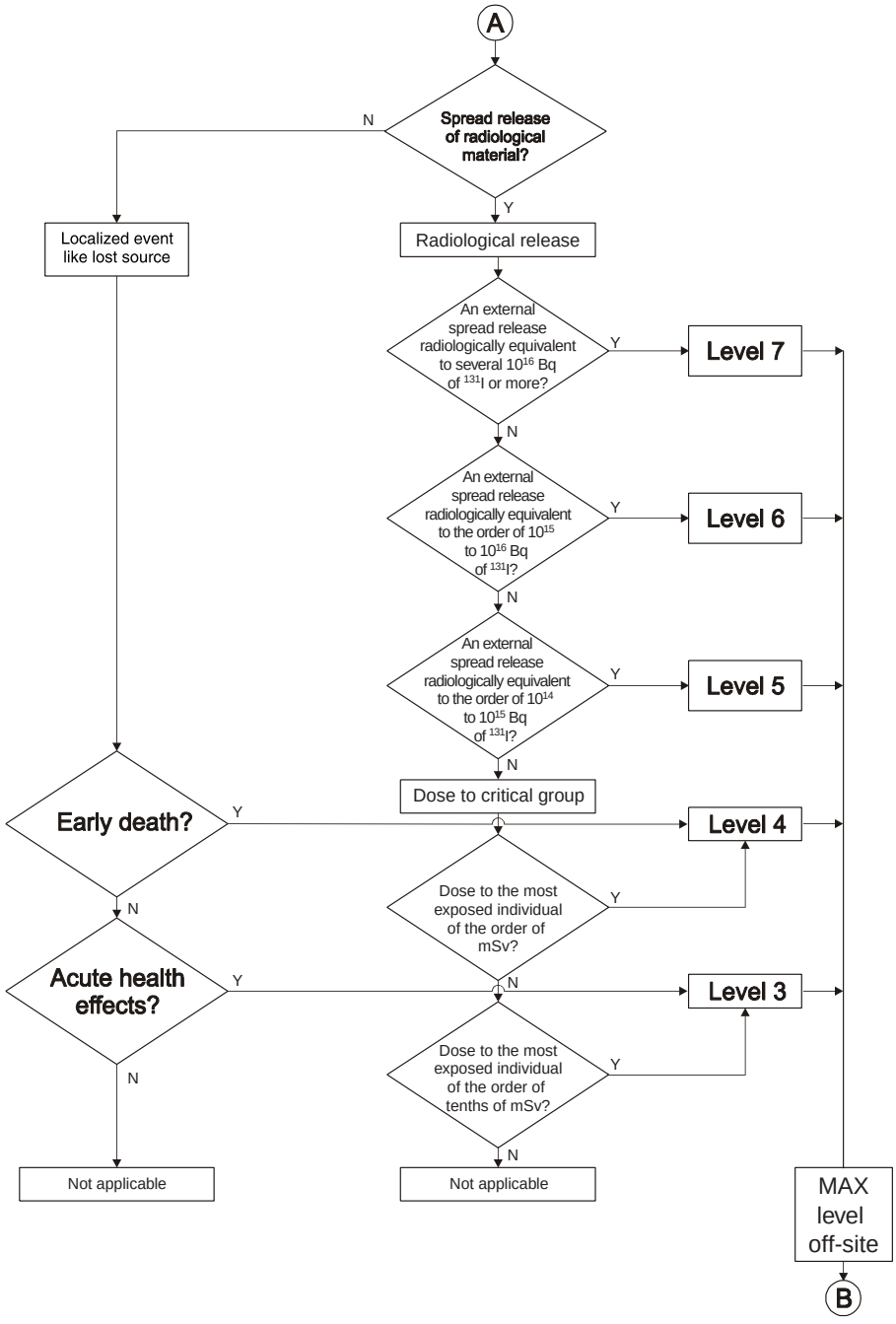
(a) Events rated at level 2 and above,
(b) Events attracting international public interest.

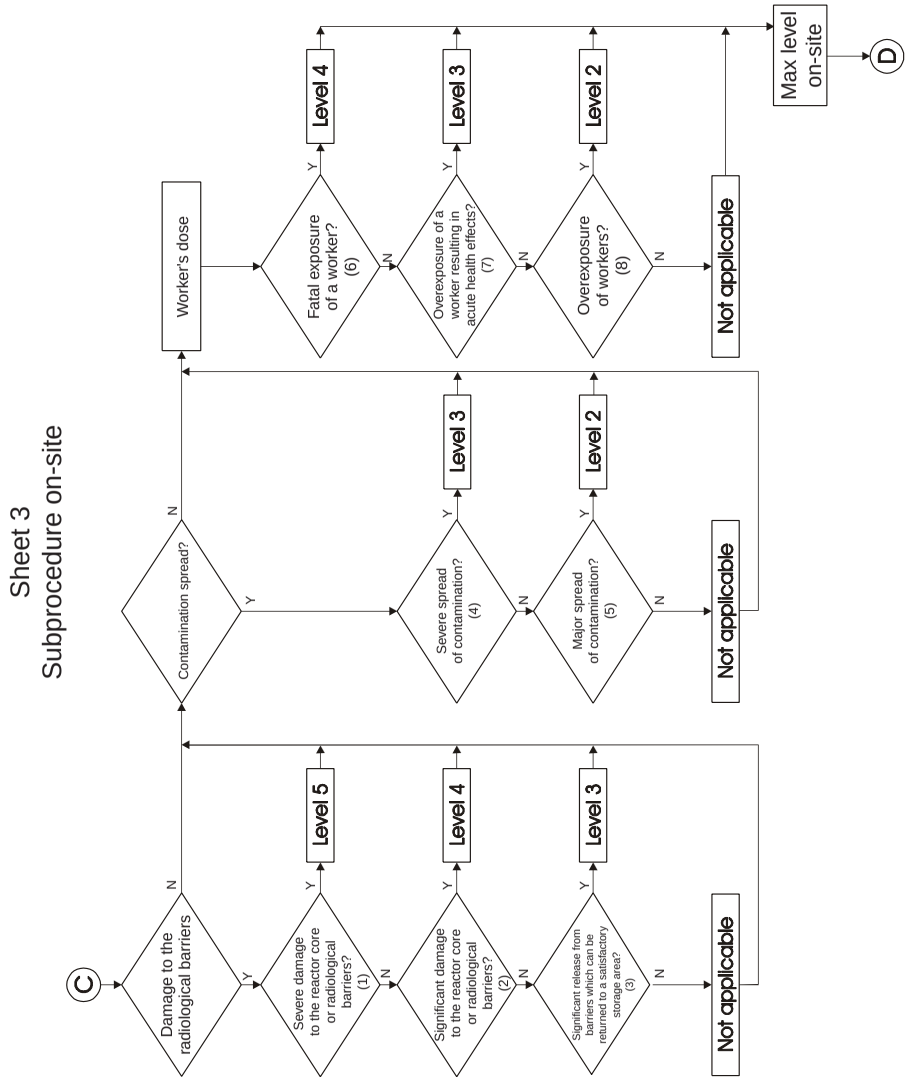
The information is presented in a specific format using the 'Event Rating Form' available from the IAEA. This form is forwarded to the IAEA INES Information Service through two redundant channels, fax machine and electronic mail. The INES Information Service is always in operation and can therefore ensure dissemination of the form at any time.

Sheet 1
INES rating procedures



Sheet 2
Subprocedure off-site




Sheet 3
Subprocedure on-site
C
Damage to the radiological barriers
N
Y
Severe damage to the reactor core or radiological barriers? (1)
Y
N
Level 5
Significant damage to the reactor core or radiological barriers? (2)
Y
N
Level 4
Significant release from barriers which can be returned to a satisfactory storage area? (3)
Y
N
Level 3
Not applicable
Contamination spread?
N
Y
Severe spread of contamination? (4)
Y
N
Level 3
Major spread of contamination? (5)
Y
N
Level 2
Not applicable
Worker's dose
Fatal exposure of a worker? (6)
Y
N
Level 4
Overexposure of a worker resulting in acute health effects? (7)
Y
N
Level 3
Overexposure of workers? (8)
Y
N
Level 2
Not applicable
Max level on-site
D

Notes for Sheet 3

1. More than a few per cent of the fuel in a power reactor is molten or more than a few per cent of the core inventory has been released from the fuel assemblies. Incidents at other installations involving a major release of radioactivity on the site (comparable with the release from core melt) with a serious off-site radiological safety threat.

2. Any fuel melting has occurred or more than about 0.1% of the core inventory of a power reactor has been released from the fuel assemblies. Events at non-reactor installations involving the release of a few thousand terabecquerels of activity from their primary containment which cannot be returned to a satisfactory storage area.

3. Events resulting in a release of a few thousand terabecquerels of activity into a secondary containment where the material can be returned to a satisfactory storage area.

4. Events resulting in a dose rate or contamination level which could easily have resulted in one or more workers receiving a dose leading to acute health effects (such as whole body exposure of the order of 1 Gy and body surface exposures of the order of 10 Gy).

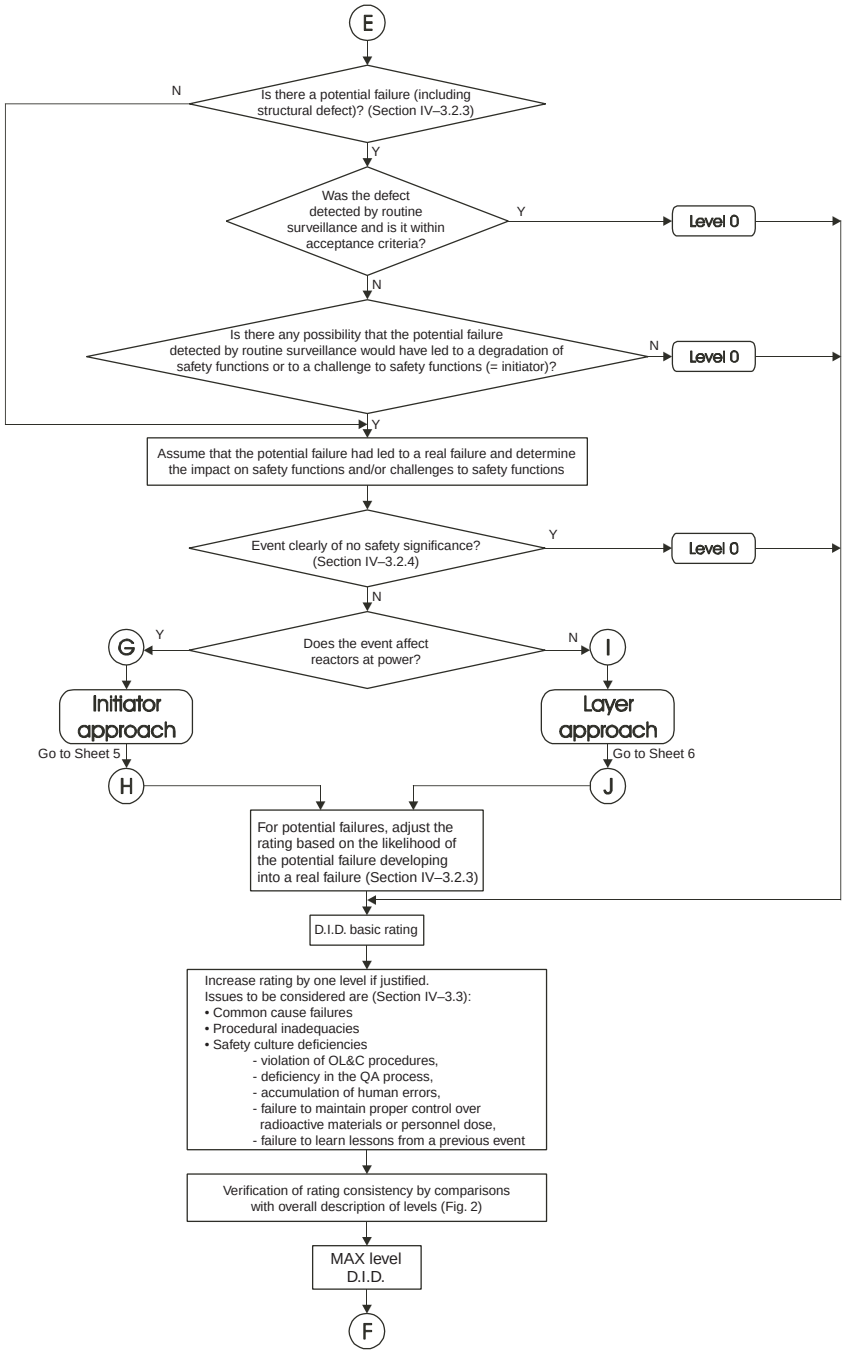
5. An event resulting in the sum of gamma plus neutron dose rates of greater than 50 mSv per hour in a plant operating area (dose rate measured 1 m from the source). An event leading to the presence of significant quantities of radioactivity in the installation, in areas not expected by design (see Section III–2.3) and which requires corrective action. In this context, "significant quantity" should be interpreted as: (a) contamination by liquids involving a total activity radiologically equivalent to a few hundred gigabecquerels of ^{106}Ru; (b) a spillage of solid radioactive material of radiological significance equivalent to the order of a few hundred gigabecquerels of ^{106}Ru, providing the surface and airborne contamination levels exceed ten times those permitted for controlled areas; (c) a release of airborne radioactive material, contained within a building and involving quantities of radiological significance equivalent to the order of a few tens of gigabecquerels of ^{131}I.

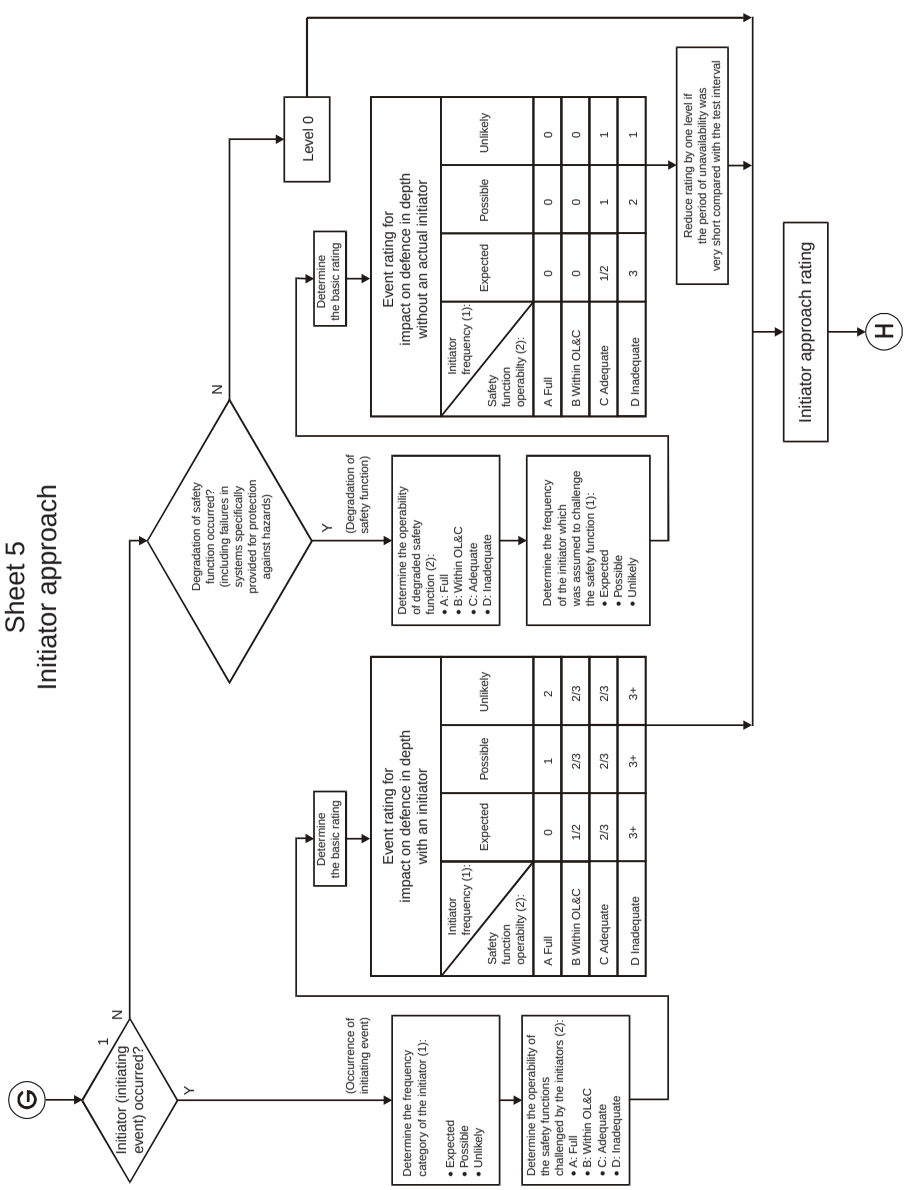
6. External irradiation of one or more workers, which results in an overexposure where a high probability of early death occurs (about 5 Gy).

7. Events resulting in a dose rate or contamination level which resulted in one or more workers receiving a dose leading to acute health effects (such as whole body exposures of the order of 1 Gy and body surface exposures of the order of 10 Gy).

8. An event resulting in a dose to one or more workers exceeding an International Commission for Radiological Protection annual dose limit for radiation workers. An event resulting in the need for significant surgery to prevent a dose that would otherwise have been about an order of magnitude above the annual dose limit.

Sheet 4
Subprocedure defence in depth (D.I.D.)




Sheet 5
Initiator approach
G
Initiator (initiating event) occurred?
1
N
Y
(Occurrence of initiating event)
Determine the frequency category of the initiator (1):
• Expected
• Possible
• Unlikely
Determine the operability of the safety functions challenged by the initiators (2):
• A: Full
• B: Within OL&C
• C: Adequate
• D: Inadequate
Determine the basic rating
Event rating for impact on defence in depth with an initiator
Initiator frequency (1): / Safety function operability (2):
Expected
Possible
Unlikely
A Full 0 1 2
B Within OL&C 1/2 2/3 2/3
C Adequate 2/3 2/3 2/3
D Inadequate 3+ 3+ 3+
Degradation of safety function occurred? (including failures in systems specifically provided for protection against hazards)
Y
(Degradation of safety function)
Determine the operability of degraded safety function (2):
• A: Full
• B: Within OL&C
• C: Adequate
• D: Inadequate
Determine the frequency of the initiator which was assumed to challenge the safety function (1):
• Expected
• Possible
• Unlikely
N
Level 0
Determine the basic rating
Event rating for impact on defence in depth without an actual initiator
Initiator frequency (1): / Safety function operability (2):
Expected
Possible
Unlikely
A Full 0 0 0
B Within OL&C 0 0 0
C Adequate 1/2 1 1
D Inadequate 3 2 1
Reduce rating by one level if the period of unavailability was very short compared with the test interval
Initiator approach rating
H

Notes for Sheet 5

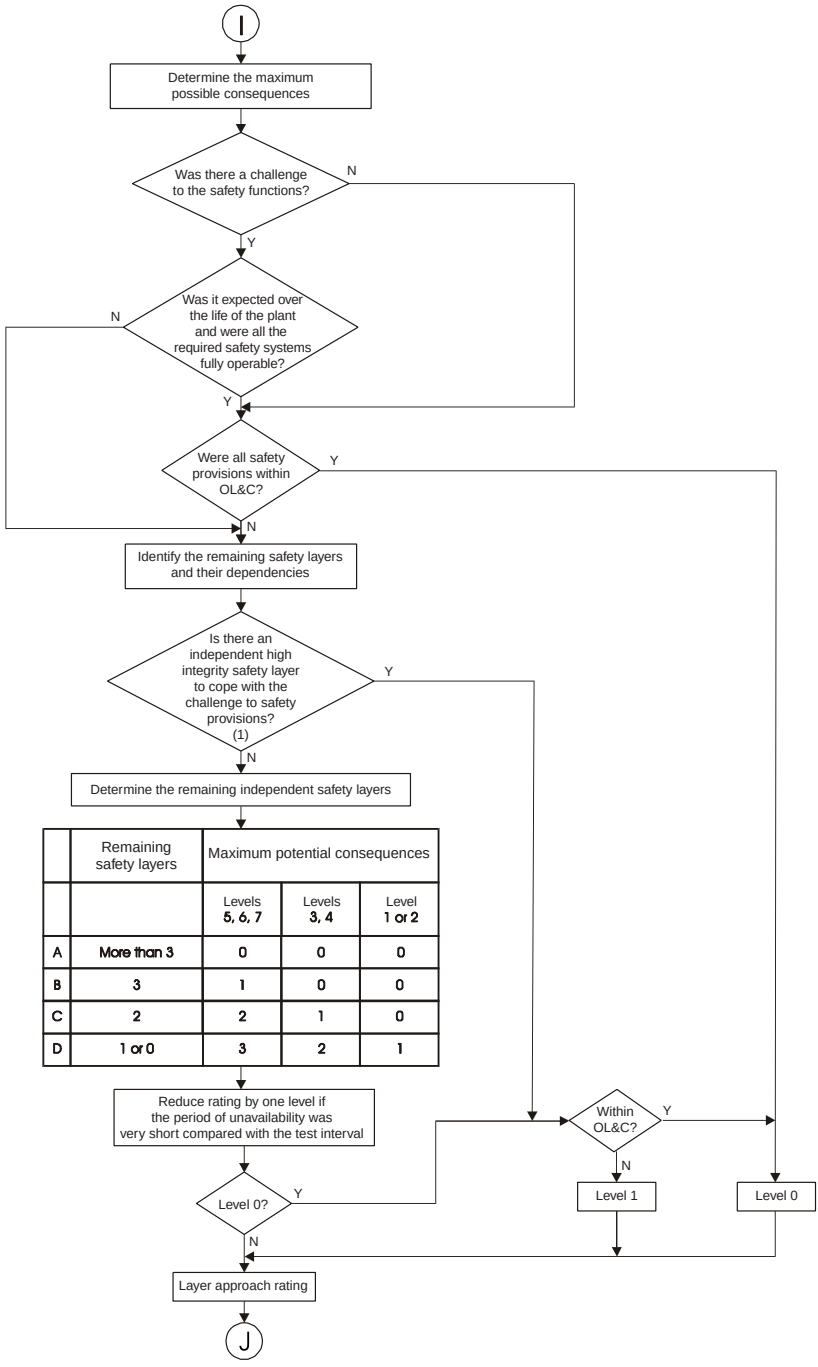
1. Definition of initiator and initiator frequency: An initiator is an occurrence that challenges the safety systems and requires them to function. In practice, the initiator may be different from the occurrence which starts the event. Frequency categories of the initiators are as follows:

   - Expected: initiators which are expected to occur once or several times during the life of the plant.
   - Possible: initiators which are not 'expected', but have an anticipated frequency during the plant lifetime of greater than about 1% (i.e. about 3×10^{-4}/a).
   - Unlikely: initiators considered in the design of the plant which are less likely than the above.

2. Safety function operability: The three basic safety functions are: (a) controlling the reactivity or the process conditions; (b) cooling the radioactive material; (c) confining the radioactive material. The function is achieved by safety systems, including support systems such as electrical supplies, cooling and instrument supplies. To provide a framework for rating events, four levels of operability are considered:

   A — Full: all safety systems and components provided by the design to cope with the particular initiator are fully operable.
   B — Minimum required (by operational limits and conditions (OL&C)): minimum operability of safety systems specified in the OL&C for continued operation at power, even for a limited time.
   C — Adequate: a level of operability of safety systems sufficient to achieve the particular safety function for the initiator being considered.
   D — Inadequate: the degraded operability of the safety systems is such that the safety function cannot be fulfilled.

Sheet 6
Layers approach



	Remaining safety layers	Maximum potential consequences		
		Levels 5, 6, 7	Levels 3, 4	Level 1 or 2
A	More than 3	0	0	0
B	3	1	0	0
C	2	2	1	0
D	1 or 0	3	2	1

Notes for Sheet 6

1. A high integrity safety layer should have all of the following characteristics:

   (a) The safety layer is designed to cope with all relevant design basis faults and is explicitly or implicitly recognized in the plant safety justification as requiring a particularly high level of reliability or integrity.
   (b) The integrity of the safety layer is assured through appropriate monitoring or inspection such that any degradation of integrity is identified.
   (c) If any degradation of the layer is detected, there are clear means of coping with the event and of implementing corrective actions, either through pre-determined procedures or through long times being available to mitigate the fault.

Part III

OFF-SITE AND ON-SITE IMPACT

III–1. OFF-SITE IMPACT

III–1.1. General description

The rating of events in terms of the off-site impact takes account of the actual radiological impact outside the site of the nuclear installation. This can be expressed in terms of the amount of activity released from a facility or the assessed dose to members of the public. It is accepted that for a significant accident at a facility, it will not be possible to determine with accuracy at an early stage the size of the off-site release. However, it should be possible to indicate the release in broad terms and thus to assign the accident to a tentative level on the scale. It is possible that subsequent re-evaluation of the extent of the release would necessitate revision of the initial estimate of the rating of the event on the scale.

It is important to note that the extent of emergency response to accidents is not used as a basis for rating. Details of the planning against accidents at nuclear plants vary from one country to another and it is also possible that precautionary measures may be taken in some cases even where they are not fully justified by the actual size of the release. For these reasons, it is the size of the release and the assessed dose which should be used to rate the event on the scale and not the protective actions taken in response to emergency plans.

Five levels have been selected, starting from level 7, where a large fraction of the core inventory of a commercial nuclear power plant is released, down to level 3, where the dose to a member of the public is numerically equivalent to about one tenth of the annual dose limit. For levels 3 and 4, the committed dose to the critical group is used to assess the appropriate level. For levels 5–7, the definitions are in terms of a quantity of activity released, radiologically equivalent to a given number of terabecquerels of ^{131}I. The reason for the change is that for these larger releases the actual dose received will depend very much on the countermeasures implemented.

The release levels were set on the basis that, taking account of the likely countermeasures, it was estimated that a level 5 release could give doses of the order of ten times the doses defined for level 4. Of course, the actual quantity of radioactivity release corresponding to the threshold for level 5 is significantly more than an order of magnitude greater than the minimum release size that would correspond to a level 4 accident.

Below level 3, off-site impact is considered as being insignificant for the purpose of rating an event on the scale. Only the on-site impact and the impact on defence in depth have to be considered at these lower levels.

Events considered under off-site impact will be of two types, both of which are considered in the definition given below. The first relates to releases that will be dispersed significantly so that the doses will be small but to a significant number of members of the public. The second refers to doses, such as could occur from a lost source or a transport event, that may be larger but to a much smaller number of people. Specific guidance is given for this latter type of event in the definitions for levels 3 and 4. The definitions of levels 5–7 apply to both types of events.

III–1.2. Definition of levels

Level 7. Major release

Definition: **An external release corresponding to a quantity of radioactivity radiologically equivalent[1] to a release to the atmosphere of several tens of thousands of terabecquerels of ^{131}I or more.**

This corresponds to the release of a large fraction of the core inventory of a power reactor, typically involving a mixture of short and long lived radioactive fission products. With such a release, there is a possibility of acute health effects. Delayed health effects over a wide area, perhaps involving more than one country, are expected. Long term environmental consequences are also likely.

Level 6. Significant release

Definition: **An external release corresponding to a quantity of radioactivity radiologically equivalent (see footnote 1) to a release to the atmosphere of the order of thousands to tens of thousands of terabecquerels of ^{131}I.**

With such a release it is very likely that protective measures such as sheltering and evacuation will be judged to be necessary to limit health effects on members of the public over the emergency planning zone.

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[1] Radiological equivalence is defined in Section III–1.3.

Level 5. Limited release

Definition: **An external release, corresponding to a quantity of radioactivity radiologically equivalent (see footnote 1) to a release to the atmosphere of the order of hundreds to thousands of terabecquerels of ^{131}I.**

As a result of the actual release, some protective measures will probably be required, for example, localized sheltering and/or evacuation to minimize the likelihood of health effects.

Level 4. Minor release

Definition: **An external release of radioactivity resulting in a dose (as defined in Section III–1.3) to the critical group of the order of a few millisieverts or an event, such as a lost source or transport event, which results in a dose to a member of the public of greater than 5 Gy (i.e. one with a high probability of early death).**

As a result of the actual release, off-site protective actions are generally unlikely, except for possible local food controls. Other actions can nevertheless be taken as a precaution against further degradation of the plant's status. Plant status is taken into account in the other areas of impact (on-site impact and impact on defence in depth).

Level 3. Very small release

Definition: **An external release of radioactivity resulting in a dose (as defined in Section III–1.3) to the critical group of the order of tenths of a millisievert or an event, such as a lost source or transport event, which results in a dose to a member of the public leading to acute health effects (such as whole body exposure of the order of 1 Gy and body surface exposure of the order of 10 Gy).**

Following such an actual release, off-site protection measures are not needed. Such measures can nevertheless be taken as a precaution against further degradation of the plant's status. Plant status is taken into account in the other areas of impact (on-site impact and impact on defence in depth).

III–1.3. Calculation of radiological equivalence and dose

For levels 5–7, food banning is likely to be implemented and therefore the relative radiological significance of a release to the atmosphere should be assessed by

comparing the total committed effective dose from all nuclides resulting from inhalation, from the external dose from the passage of the cloud of active material and from the long term external irradiation of deposited activity, i.e. from all pathways except ingestion. Using the assumptions given in Appendix I, the multiplication factor for a range of isotopes has been calculated and is given in Table I. The actual activity released should be multiplied by the factor given and then compared with the values given in the definition of each level.

For levels 3 and 4, there is likely to be little or no food banning, the relative radiological significance is assessed by comparing the committed effective dose for intakes by all routes to the critical group. This should be calculated using the standard national assumptions for dose assessment without taking account of the wind direction at the time of the release or the time of year at which the release occurred. It is not possible to give multiplication factors for levels 3 and 4 as the dose via ingestion will depend on the local agricultural practices.

TABLE I. RADIOLOGICAL EQUIVALENCE FOR OFF-SITE IMPACT (*this applies to levels 5–7 only*)

Isotope	Multiplication factor
^{3}H	0.02
^{131}I	1
^{137}Cs	30
^{134}Cs	20
^{132}Te	0.3
^{54}Mn	4
^{60}Co	50
^{90}Sr	10
^{106}Ru	7
$^{235}U(S)$[a]	800
$^{235}U(M)$[a]	300
$^{235}U(F)$[a]	100
$^{238}U(S)$[a]	700
$^{238}U(M)$[a]	300
$^{238}U(F)$[a]	50
U_{nat}	800
^{239}Pu (Class Y)	10 000
^{241}Am	9000
Noble gases	Negligible (effectively 0)

[a] Lung absorption types: S — slow; M — medium; F — fast. If unsure, use the most conservative value.

Liquid discharges resulting in critical group doses significantly higher than that appropriate for level 4 would need to be rated at level 5 or above but again, the assessment of radiological equivalence would be site specific and therefore detailed guidance cannot be provided here.

III–2. ON-SITE IMPACT

III–2.1. General description

The rating of events under on-site impact takes account of the actual impact within the site of the nuclear installation, regardless of the possible off-site releases and defence in depth implications. It considers the extent of major radiological damage, for example core damage, the spread of radioactive products within the site but outside their as-designed containments and the levels of doses to workers.

Events resulting in radiological damage are rated at levels 4 and 5, events resulting in contamination are rated at levels 2 and 3 and events resulting in high doses to workers are rated at levels 2–4. The significance of contamination is measured either by the quantity spread or the resultant dose rate. These criteria relate to dose rates in an operating area but do not require that a worker was actually present. They should not be confused with the criteria for doses to workers which relate to doses actually received.

It is accepted that the exact nature of damage to plant may not be known for some time following an accident with on-site consequences of this nature. However, it should be possible to estimate in broad terms the likelihood of major or minor damage and to decide whether to rate an event provisionally at level 4 or 5 on the scale. It is possible that subsequent re-evaluation of the state of the plant would necessitate re-rating of the event.

Below level 2, on-site impact is considered as insignificant for the purpose of rating an event on the scale; it is only the impact on defence in depth which has to be considered at these lower levels.

III–2.2. Definition of levels

Level 5. Severe damage to the reactor core or radiological barriers

Definition: **More than a few per cent of the fuel in a power reactor is molten or more than a few per cent of the core inventory has been released from the fuel assemblies. Incidents at other installations involving a major release of radioactivity on the site (comparable with the release from a core melt) with a serious off-site radiological safety threat.**

Examples of non-reactor accidents would be a major criticality accident, or a major fire or explosion releasing large quantities of activity within the installation.

Level 4. Significant damage to the reactor core or radiological barriers or fatal exposure of a worker

Definition: **Any fuel melting has occurred or more than about 0.1% of the core inventory of a power reactor has been released from the fuel assemblies.**
Events at non-reactor installations involving the release of a few thousand terabecquerels of activity from their primary containment[2] which cannot be returned to a satisfactory storage area.
External irradiation of one or more workers, which results in a dose greater than 5 Gy (i.e. one with a high probability of early death).

Level 3. Severe spread of contamination and/or overexposure of a worker resulting in acute health effects

Definition: **Events resulting in a dose rate or a contamination level which did or easily could have resulted in one or more workers receiving a dose leading to acute health effects (such as whole body exposures of the order of 1 Gy and body surface exposures of the order of 10 Gy).[3]**
Events resulting in the release of a few thousand terabecquerels of activity into a secondary containment (see footnote 2) where the material can be returned to a satisfactory storage area.

Level 2. Major spread of contamination and/or overexposure of workers

Definition: **Events resulting in a dose to one or more workers exceeding a statutory annual dose limit for radiation workers.**
Events resulting in the sum of gamma plus neutron dose rates of greater than 50 mSv per hour in a plant operating area (dose rate measured 1 m from the source).

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[2] In this context, the terms primary and secondary containment refer to the containment of radioactive materials at non-reactor installations and should not be confused with the similar terms used for reactor containments.

[3] This requires a judgement based on dose rate, time and protective measures.

Events leading to the presence of significant quantities of radioactivity in the installation, in areas not expected by design (see the definitions at the end of Part IV) and which require corrective action. In this context 'significant quantity' should be interpreted as:

(a) Contamination by liquids involving a total activity radiologically equivalent to a few hundred gigabecquerels of ^{106}Ru.
(b) A spillage of solid radioactive material of radiological significance equivalent to the order of a few hundred gigabecquerels of ^{106}Ru, providing the surface and airborne contamination levels exceed ten times those permitted for operating areas (see the definitions at the end of Part IV).
(c) A release of airborne radioactive material, contained within a building and involving quantities of radiological significance equivalent to the order of a few tens of gigabecquerels of ^{131}I.

III–2.3. Calculation of radiological equivalence

The assumptions to be used in calculating radiological equivalence for on-site impact are given in Appendix I. On the basis of these assumptions, the multiplying factor for a range of isotopes has been calculated and is given in Table II. The actual activity released should be multiplied by the factor given and then compared with the values given in the definition of each level for either ^{131}I or ^{106}Ru.

TABLE II. RADIOLOGICAL EQUIVALENCE FOR ON-SITE IMPACT

Isotope	Multiplication factor for ^{131}I equivalence	Multiplication factor for ^{106}Ru equivalence
^{3}H	0.002	0.0006
^{131}I	1	0.3
^{137}Cs	0.6	0.2
^{134}Cs	0.9	0.3
^{132}Te	0.3	0.1
^{54}Mn	0.1	0.03
^{60}Co	1.5	0.5
^{90}Sr	7	2
^{106}Ru	3	1
$^{235}U(S)$[a]	600	700
$^{235}U(M)$[a]	200	200
$^{235}U(F)$[a]	50	20
$^{238}U(S)$[a]	500	30
$^{238}U(M)$[a]	100	170
$^{238}U(F)$[a]	50	20
U_{nat}	600	200
^{239}Pu (Class Y)	9000	3000
^{241}Am	2000	700
Noble gases	Negligible (effectively 0)	Negligible (effectively 0)

[a] Lung absorption types: S — slow; M — medium; F — fast. If unsure, use the most conservative value.

Part IV

IMPACT ON DEFENCE IN DEPTH

This part of the manual is divided into three main sections. The first gives the background to what is meant by defence in depth. This will probably be familiar to most readers. The second section gives the general principles that are to be used to rate events under defence in depth. As they need to cover a wide range of types of installations and events, they are general in nature. In order to ensure that they are applied in a consistent manner, Section 3 gives more detailed guidance. The guidance is further expanded in Part V, which gives specific guidance for certain types of events and provides a number of worked examples.

IV–1. BACKGROUND

The avoidance of radiological accidents and incidents, and hence the safety of a nuclear installation, is based on good design and operation. A defence in depth approach is generally applied to both of these aspects and allowance is made for the possibility of equipment failure, human error and the occurrence of unplanned developments.

The definition of defence in depth by the International Nuclear Safety Advisory Group is as follows:

> "To compensate for potential human and mechanical failures, a defence in depth concept is implemented, centred on several levels of protection including successive barriers preventing the release of radioactive material to the environment. The concept includes protection of the barriers by averting damage to the plant and to the barriers themselves. It includes further measures to protect the public and the environment from harm in case these barriers are not fully effective."[4]

Similar defence in depth provisions are provided at all nuclear installations and for the transport of radioactive material. They cover protection of the public and the workforce, and include the means to prevent the transfer of material into poorly

[4] INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Basic Safety Principles for Nuclear Power Plants 75-INSAG-3 Rev. 1, INSAG-12, IAEA, Vienna (1999) 17.

shielded locations as well as to prevent radioactive release. Defence in depth is, therefore, a combination of conservative design, quality assurance, surveillance activities, mitigative measures and a general safety culture that strengthens each of the successive layers.

Safe operation is maintained by the three basic safety functions:

(a) Controlling the reactivity or the process conditions,
(b) Cooling the radioactive material,
(c) Confining the radioactive material.

Each of the safety functions is assured by good design, well controlled operation and a range of systems and administrative controls. Within the safety justification for the plant, operational systems may be distinguished from safety provisions; if operational systems fail, then additional safety provisions will operate so as to maintain the safety function. Safety provisions can either be procedures, administrative controls or passive or active systems, which are usually provided in a redundant way, with their availability controlled by operational limits and conditions (OL&C).

The frequency of challenge of the safety provisions is minimized by good design, operation, maintenance, surveillance, etc. For example, the frequency of failures of the primary circuit of a reactor is minimized by design margins, quality control, operational constraints, surveillance, and so on. Similarly, the frequency of reactor transients is minimized by operational procedures, control systems, etc. Normal operational and control systems contribute to minimizing the frequency of challenges to safety provisions.

In some situations it is not possible to reduce significantly the frequency of the challenge of safety provisions, for example attempted entry into cells potentially containing sources. In these cases the safety functions are assured solely by safety provisions of appropriate integrity.

IV–2. GENERAL PRINCIPLES FOR THE RATING OF EVENTS

This guidance is for application to a wide range of nuclear installations and the radioactive inventory and time-scales of events at such installations will vary widely. These are important factors to be taken into account in rating events and it is inevitable that the guidance here is general and that judgement must be applied. More specific guidance is given in the later sections.

Although three levels for impact on defence in depth are available above level 0, for some installations the maximum possible on-site or off-site consequences are limited by the radioactive inventory and the release mechanism. Clearly the maximum possible level with respect to impact on the defence in depth, where an accident

has been prevented, should be lower than the maximum possible level with respect to on-site or off-site impact. If the maximum possible on-site or off-site level for a particular activity cannot be greater than level 4 on the scale because of the limited potential consequences, a maximum rating of level 2 is appropriate under defence in depth. Similarly, if the maximum potential level cannot exceed level 2, then the maximum under defence in depth is level 1.

One facility can, of course, cover a number of activities and each activity must be considered separately in this context. For example, waste storage and reactor operations should be considered as separate activities, even though they can both occur at one facility.

Having identified the upper limit to the rating under defence in depth, the approach to rating is based on assessing the likelihood that the event could have led to an accident, not by using probabilistic techniques directly but by considering whether safety provisions were challenged and what additional failures of safety provisions would be required to result in an accident. Consideration is also given as to whether any underlying cultural issues are evident in the event that might have increased the likelihood of the event leading to an accident.

The following steps should therefore be followed to rate an event:

(1) The upper limit to the rating under defence in depth should be established by taking account of the maximum potential radiological consequences (i.e. the maximum potential rating for the relevant activities at that facility under off-site and on-site impact). Further guidance on establishing the maximum potential consequences is given in Section IV–3.1.

(2) The basic rating should then be determined by taking account of the number and effectiveness of the safety provisions available (hardware and administrative) for prevention, surveillance and mitigation, including passive and active barriers. In identifying the number and effectiveness of such provisions it is important to take account of the time available and the time required for identifying and implementing appropriate corrective action. Further guidance on the assessment of safety provisions is provided in Section IV–3.2.

(3) In addition to the above considerations, increasing the basic rating should be considered, as explained in Section IV–3.3, within the upper limit of the defence in depth rating established in item (1) above. Uprating allows for those aspects of the event that may indicate a deeper degradation of the plant or the organizational arrangements of the facility. Factors considered are common cause failures, procedural inadequacies and safety culture deficiencies. Such factors are not included in the basic rating and may indicate that the significance of the event with respect to defence in depth is higher than the one considered in the basic rating process. Accordingly, in order to communicate the true significance of the event to the public, uprating by one level is considered.

Clearly, as well as considering the event under defence in depth, each event must also be considered against off-site and on-site impact.

IV–3. DETAILED GUIDANCE FOR RATING EVENTS

IV–3.1. Identification of maximum potential consequences

For the assessment of events affecting the majority of the reactor core or the fuel in the spent fuel pool of power reactors, it is generally not necessary to specifically consider the maximum potential consequences. The theoretical possibility of a large release is recognized and therefore the upper limit to the rating under defence in depth is level 3.

For other facilities, or for activities involving only a small fraction of the core inventory (e.g. fuel handling), it is necessary to consider the maximum potential consequences (i.e. the maximum potential rating under off-site and on-site impact) should all the safety provisions fail. For some facilities it may not be physically possible to reach the upper levels of INES even from extremely unlikely accidents. The maximum potential consequences are not specific to the type of event but apply to a set of operations at a facility.

In assessing the maximum potential rating under off-site and on-site impact, the following general principles should be taken into account:

(a) Any one site may contain a number of facilities with a range of tasks carried out at each facility. Thus the maximum potential rating should be specific to the type of facility at which the event occurred and the type of operations being undertaken at the time of the event.
(b) It is necessary to consider both the radioactive inventory that could potentially have been involved in the event, the physical and chemical properties of the material involved, and the mechanisms by which that activity could have been dispersed.
(c) The consideration should not focus on the scenarios considered in the safety justification of the plant but should consider physically possible accidents had all the plant safety provisions threatened by the event been deficient.

These principles can be illustrated by the following examples:

(1) For events associated with maintenance cell entry interlocks, the maximum potential consequences are likely to be related to worker exposure. If the radiation levels are sufficiently high to cause worker death if the cell is entered and no mitigative actions are taken, then the maximum potential rating is at level 4 under on-site impact.

(2) For events involving small research reactors (i.e. with power less than 1 MW), although the physical mechanisms exist for the dispersal of a significant fraction of the inventory (either through criticality accidents or loss of fuel cooling), the total inventory is such that the maximum potential rating could not be higher than level 4, either on-site or off-site, even if all the safety provisions fail.

(3) For reprocessing facilities and other facilities processing plutonium compounds, the inventory and physical mechanisms which exist for the dispersal of a significant fraction of that inventory (either through criticality accidents, chemical explosions or fires), are such that the maximum potential rating could exceed level 4, either under off-site or on-site impact, if all the safety provisions fail.

(4) For uranium fuel fabrication and enrichment plants, releases have chemical and radiological safety aspects. It has to be emphasized that the chemical risk posed by the toxicity of fluorine and uranium predominates over the radiological risk. INES, however, is only related to the assessment of the radiological hazard. From a radiological standpoint, no severe off-site or on-site consequences exceeding a rating of level 4 are conceivable from a release of uranium or its compounds.

IV–3.2. Identification of basic rating taking account of the effectiveness of safety provisions

Because the safety analysis for reactor installations during power operation follows a common international practice, it is possible to give more specific guidance about how to assess the safety provisions for events involving reactors at power. In addition, as noted at the start of Section IV–3.1, the rating does not need to explicitly consider the maximum potential consequences. The approach is based on consideration of **initiators**, **safety functions** and **safety systems**. These terms will be familiar to those involved in safety analysis but further explanation of the terms is provided below. Other events at reactor sites, e.g. those associated with a shutdown reactor or with other facilities on the site, should be rated using the safety layers approach described in Section IV–3.2.2. Similarly, events involving research reactors should use the safety layers approach to take proper account of maximum potential consequences and design philosophy. An overview of the approach to help those new to the scale is given in Appendix II.

IV–3.2.1. Events occurring on reactors at power (initiator approach)

An initiator or initiating event is an identified event that leads to a deviation from the normal operating state and challenges one or more safety functions.

Initiators are used in safety analysis to evaluate the adequacy of installed safety systems: the initiator is an occurrence that challenges the safety systems and requires them to function.

Events involving an impact on the plant defence in depth will generally be of two possible forms:

— Either an initiator (initiating event) which requires the operation of some particular safety systems designed to cope with the consequences of this initiator;
— Or degraded operability of a safety function owing to the operability of one or more safety systems being degraded without the occurrence of the initiator for which the safety systems had been provided.

In the first case, the event rating depends mainly on the extent to which the operability of the safety function is degraded. However, the severity also depends on the anticipated frequency of the particular initiator.

In the second case, no deviation from normal operation of the plant actually occurs, but the observed degradation of the operability of the safety function could have led to significant consequences if one of the initiators for which the degraded safety systems are provided had actually occurred. In such a case, the event rating again depends on:

— The anticipated frequency of the potential initiator,
— The operability of the associated safety function assured by the operability of particular safety systems.

It has to be pointed out that one particular event could be categorized under both cases.

The basic approach to rating such events is therefore to identify the frequency of the relevant initiators and the operability of the affected safety functions. Two tables are then used to identify the appropriate basic rating. Further information on the derivation of the tables is given in Appendix III. Detailed guidance on rating is given below.

IV–3.2.1.1. Identification of initiator frequency

Four different frequency categories have been selected:

(1) *Expected.* This covers initiators expected to occur once or several times during the operating life of the plant.
(2) *Possible.* Initiators which are not ‘expected’, but have an anticipated frequency during the plant lifetime of greater than about 1% (i.e. about 3×10^{-4} per year).

(3) *Unlikely.* Initiators considered in the design of the plant which are less likely than the above.
(4) *Beyond design.* Initiators of very low frequency, not normally included in the conventional safety analysis of the plant. When protection systems are introduced against these initiators, they do not necessarily include the same level of redundancy or diversity as measures against design basis accidents.

Each plant has its own list and classification of initiators. Typical examples of design basis initiators categorized into the previous classes are given in Appendix IV. Small plant perturbations that are corrected by control (as opposed to safety) systems are not included in the initiators. The initiator may be different from the occurrence which starts the event; on the other hand a number of different event sequences can often be grouped under a single initiator.

For many events, it will be necessary to consider more than one initiator, each of which will lead to a rating. The event level will be the highest of the levels associated with each initiator. For example, a power excursion in a reactor could be an initiator challenging the protection function. Successful operation of the protection system would then lead to a shutdown. It would then be necessary to consider the reactor trip as an initiator challenging the fuel cooling function.

IV–3.2.1.2. Safety function operability

The three basic safety functions are:

(a) Controlling the reactivity or the process conditions,
(b) Cooling the radioactive material,
(c) Confining the radioactive material.

These functions are provided by passive systems (such as physical barriers) and active systems (such as the reactor protection system). Several safety systems may contribute to a particular safety function, and the function may still be achieved even with one system unavailable. Equally, support systems such as electrical supplies, cooling and instrument supplies will be required to ensure that a safety function is achieved. It is important that it is the operability of the safety function that is considered when rating events, not the operability of an individual system. A system or component shall be considered operable when it is capable of performing its required function in the required manner.

Operational limits and conditions govern the operability of each safety system. In most countries they are included within the Technical Specifications.

The operability of a safety function for a particular initiator can range from a state where all the components of the safety systems provided to fulfil that function

are fully operable to a state where the operability is insufficient for the safety function to be achieved. To provide a framework for rating events, four categories of operability are considered.

A. Full

All safety systems and components which are provided by the design to cope with the particular initiator in order to limit its consequences are fully operable (i.e. redundancy/diversity is available).

B. Minimum required by OL&C

The minimum operability of safety systems providing the required safety function specified in OL&C for which continued operation at power is permitted, even for a limited time. This level of operability will generally correspond to the minimum operability of the different safety systems for which the safety function can be achieved for all the initiators considered in the design of the plant. However, for certain particular initiators redundancy and diversity may still exist.

C. Adequate

A level of operability of safety systems sufficient to achieve the particular safety function for the initiator being considered. For some safety systems, this will correspond to a level of operability lower than that required by OL&C. An example would be where diverse safety systems are each required to be operable by OL&C, but only one is operable, or where all safety systems which are designed to assure a safety function are inoperable for such a short time that the safety function, although outside OL&C, is still assured by other means (for example, the safety function 'cooling of the fuel' may be assured if a total station blackout occurs for only a short time). In other cases, categories B and C may be the same.

D. Inadequate

The degraded operability of the safety systems is such that the safety function cannot be fulfilled for the initiator being considered.

It should be noted that although C and D represent a range of plant states, A and B represent specific operabilities. Thus the actual operability may be between that defined by A and B, i.e. the operability may be less than full but more than the minimum allowed for continued operation at power. This is considered in Section IV–3.2.1.3(a).

IV–3.2.1.3. Assessment of the basic rating

In order to obtain a basic categorization, first decide whether there was an actual challenge to the safety systems (a real initiator). If so, then Section IV–3.2.1.3(a) is appropriate, otherwise Section IV–3.2.1.3(b) is appropriate. It may be necessary to consider an event using both sections if an initiator occurs and reveals a reduced operability in a function not challenged by the real initiator, e.g. if a reactor trip without loss of off-site power reveals a reduced operability of diesels. For events involving potential failures, e.g. discovery of structural defects, a similar approach is used as described in Section IV–3.2.3.

(a) Events with a real initiator

The first step is to decide the frequency with which that type of initiator was expected by design. In deciding the appropriate category, it is the frequency that was assumed in the safety case (the justification of the safety of the plant and its operating envelope) for the plant that is relevant. Appendix IV provides some examples.

The second step is to determine the operability of the safety functions challenged by the initiator. It is important that only those safety functions challenged are considered. If the degradation of other safety systems is discovered, it should be assessed using Section IV–3.2.1.3(b) against the initiator that would have challenged that safety function. It is also important to note that in deciding whether the operability is within OL&C, it is the operability requirements prior to the event that must be considered, not those that apply during the event. If the operability is within OL&C but also just adequate, category C should be used.

The event rating should then be determined from Table III. Where a choice of rating is given, the choice should be based on the extent of redundancy and diversity

TABLE III. EVENTS WITH A REAL INITIATOR

Initiator frequency / Safety function operability		Expected	Possible	Unlikely
A	Full	0	1	2
B	Within OL&C	1/2	2/3	2/3
C	Adequate	2/3	2/3	2/3
D	Inadequate	3+	3+	3+

available for the initiator being considered. If the safety function operability is just adequate (i.e. one further failure would have lead to an accident), level 3 is appropriate. In cell B1 of Table III, the lower value would be appropriate if there is still considerable redundancy and/or diversity available.

Where the safety function operability is greater than the minimum required by OL&C, but less than 'Full', there may be considerable redundancy and diversity available for expected initiators. In such cases, level 0 would be more appropriate.

Beyond design initiators are not included specifically in Table III. If such an initiator occurs, then levels 2 or 3 are appropriate under defence in depth depending on the redundancy of the systems providing protection. However, it is possible that beyond design initiators will lead to an accident requiring classification under off-site or on-site impacts.

The occurrence of internal and external hazards such as fires, external explosions or tornados may be rated using the table. The hazard itself should not be considered as the initiator, but the safety systems that remain operable should be assessed against an initiator that occurred and/or against potential initiators.

(b) Events without a real initiator

The first step is to determine the safety function operability. In practice, safety systems or components may be in a state not fully described by any of the four categories. The operability may be less than full but more than the minimum required by OL&C, or the whole system may be available but degraded by loss of indications. In such cases the relevant categories should be used to give the possible range of the rating, and judgement used to determine the appropriate rating. If the operability is just adequate but still within OL&C, category B should be used.

The second step is to determine the frequency of the initiator for which the safety function is required. If there is more than one relevant initiator, then each must be considered. The one giving the highest rating should be used. If the frequency lies on the boundary between two categories some judgement will need to be applied. For systems specifically provided for protection against hazards, the hazard should be considered as the initiator.

The event rating should then be determined from Table IV. Where a choice of rating is given, the choice should be based on whether the operability is just adequate or whether redundancy and/or diversity still exists for the initiator being considered. If the period of inoperability was very short compared with the interval between tests of the components of the safety system, consideration should be given to reducing the basic rating of the event.

Beyond design initiators are not included specifically in Table IV. Where the operability of the affected safety function is less than the minimum required by

TABLE IV. EVENTS WITHOUT A REAL INITIATOR

Initiator frequency / Safety function operability		Expected	Possible	Unlikely
A	Full	0	0	0
B	Within OL&C	0	0	0
C	Adequate	1/2	1	1
D	Inadequate	3	2	1

OL&C, level 1 is appropriate. If the operability is greater than the minimum required by OL&C, or OL&C do not provide any limitations on the system operability, level 0 is appropriate.

IV–3.2.2. All other events, i.e. any event not associated with reactors at power (the layers approach)

To rate an event, it is necessary to consider the safety provisions and assess the number of separate safety layers that prevented an accident. In doing so it is also necessary to consider the time available and the time required to take effective corrective action. Each of these aspects is considered below.

IV–3.2.2.1. Time available

In some situations, the time available to carry out corrective actions may be significantly greater than the time required for those actions and may therefore allow additional safety layers to be made available. These additional safety layers may be taken into account provided that procedures exist for carrying out the required actions. In some cases, the time available may be such that there are a whole range of potential safety layers that can be made available and it has not been considered necessary in the safety justification to identify each of them in detail or to include in the procedure the detail of how to make each of them available. In such cases this long time available provides a highly reliable safety layer and this must also be taken into account, as explained in the next section.

IV–3.2.2.2. Identification of safety layers

A safety layer should be considered as a safety provision that cannot be broken down into redundant parts. Thus, if the cooling function was provided by two separate 100% trains, it should be considered as two separate safety layers, unless they have a common non-redundant support system.

Safety layers can be based on passive design, active components or administrative controls. They can include surveillance procedures, though it should be noted that surveillance alone does not provide a safety layer; the means to implement corrective action are also required.

When considering the number of safety layers it is necessary to ensure that the effectiveness of a number of separate hardware layers is not reduced by a common support system or a common operator action in response to alarms or indications. In such cases, although there may be several hardware layers, there may be only one effective safety layer.

When considering administrative controls as safety layers it is important to check the extent to which separate procedures can be considered independent and to check that the procedure is of sufficient reliability to be regarded as a safety layer. It is not possible to give more explicit guidance, and inevitably judgement must be used.

In some situations, a high integrity safety layer may be available, for example a properly transported fuel transport flask, a reactor pressure vessel or a safety provision based on naturally occurring passive phenomena such as convective cooling. In such cases as the layer is demonstrated to be of extremely high integrity/reliability, it would clearly be inappropriate to only consider it as a single safety layer when applying this guidance. A high integrity safety layer should have the following characteristics:

(a) The safety layer is designed to cope with all relevant design basis faults and is explicitly or implicitly recognized in the plant safety justification as requiring a particularly high level of reliability or integrity;
(b) The integrity of the safety layer is assured through appropriate monitoring or inspection such that any degradation of integrity is identified;
(c) If any degradation of the layer is detected, there are clear means of coping with the event and of implementing corrective actions, either through pre-determined procedures or through long times being available to repair or mitigate the fault.

An example of a high integrity layer would be a vessel. Administrative controls would not normally meet the requirements of a high integrity layer though, as noted above, certain operating procedures can also be regarded as high integrity

safety layers if there are very long time-scales available to perform the actions required, and to correct operator errors should they occur, and there are a wide range of available actions.

IV–3.2.2.3. Assessment of the basic rating

Having identified the maximum potential consequences and the number of effective safety layers, the basic rating should be determined as follows:

(1) The safety analysis for the plant will identify a wide range of events that have been taken into account in the design. It will recognize that some of these could reasonably be expected to occur over the life of the plant (i.e. they will have a frequency greater than 1/*N* per year, where *N* is the expected plant life). If the challenge to the safety provisions that occurred in the event was such that an expected event and the safety systems provided to cope with that event were fully available before the event and behave as expected, the event should be rated at level 0. Similarly, if no actual challenge to the safety provisions occurs but they are discovered to be degraded, the event should be rated at level 0 if the degraded operability of the safety provisions was still within OL&C.
(2) For all other situations, Table V should be used to determine the basic rating.

If only one safety layer remains but that layer meets all the requirements of a high integrity safety layer outlined above, a basic rating of level 0 would be more appropriate.[5]

If the period of unavailability of a safety layer was very short compared with the interval between tests of the components of the safety layer, consideration should be given to reducing the basic rating of the event. This approach inevitably requires more judgement than that described in Section IV–3.2.1, but Section V–1 gives guidance for specific types of events and Section V–2 provides some generalized examples of the use of the safety layers approach.

IV–3.2.3. Potential events (including structural defects)

Some events do not of themselves challenge the safety provisions but do correspond to an increased likelihood of a challenge. Examples are the discovery of

[5] If the operability of safety layers was outside the requirements of OL&C, the guidance in Section IV–3.3 may lead to a rating of level 1.

TABLE V. RATING EVENTS USING THE LAYERS APPROACH

Maximum potential consequences / No. of remaining safety layers		INES levels 5, 6, 7	INES levels 3, 4	INES levels 2 or 1
A	More than 3	0[a]	0[a]	0[a]
B	3	1	0[a]	0[a]
C	2	2	1	0[a]
D	1 or 0	3	2	1

[a] If the operability of safety layers was outside the requirements of OL&C, the guidance in Section IV–3.3 may lead to a rating of level 1.

structural defects, a leak terminated by operator action or faults discovered in process control systems. The approach to rating such events is described below.

The surveillance programme is intended to identify structural defects before their size becomes unacceptable. If the defect is within this size, then level 0 would be appropriate. If the defect is larger than expected under the surveillance programme, categorization of the defects needs to take account of two factors.

First, the safety significance of the defective component should be determined by assuming that the defect had led to failure of the component and applying the appropriate part of Section IV–3. If using Section IV–3.2.1 (reactors at power), then if the defect is in a safety system, applying Section IV–3.2.1.3(b) will give the upper limit of the basic rating. The possibility of common mode failure may need to be considered. If the defect was in a component whose failure could have led to an initiator, then applying Section IV–3.2.1.3(a) will give the upper value of the basic rating.

The potential rating derived in this way should then be adjusted depending on the likelihood that the defect would have led to component failure, and by consideration of the additional factors discussed in Section IV–3.3.

Other potential events can be assessed in a similar way to that described above. First, the significance of the potential challenge should be evaluated by assuming that it had actually occurred and applying the appropriate part of Section IV–3, based on the operability of safety provisions that existed at the time. Secondly, the rating should be reduced, depending on the likelihood that the potential challenge could have developed from the event that actually occurred. The level to which the rating should be reduced must be based on judgement.

IV–3.2.4. Events rated below scale at level 0

In general, events should be classified below scale at level 0 only if application of the procedures described above does not lead to a higher rating. However, provided none of the additional factors discussed in Section IV–3.3 are applicable, the following types of event are typical of those that will be categorized as below scale at level 0:

— Reactor trip proceeding normally;
— Spurious operation of the safety systems[6] followed by normal return to operation without affecting the safety of the installation;
— No significant degradation of the barriers (leak rate less than OL&C);
— Single failures or component inoperability in a redundant system discovered during scheduled periodic inspection or test.

IV–3.3. Consideration of additional factors

Particular aspects may challenge simultaneously different layers of the defence in depth and are consequently to be considered as additional factors which may justify an event having to be classified one level above the one resulting from the previous guidance.

The main additional factors which act in such a way are:

— Common cause failures,
— Procedural inadequacies,
— Safety culture deficiencies.

Because of such factors, it may happen that an event could be rated at level 1, although of no safety significance on its own, without taking those additional factors into account.

[6] Spurious operation in this respect would include operation of a safety system as a result of a control system malfunction, instrument drift or individual human error. However, the actuation of the safety system initiated by variations in physical parameters which have been caused by unintended actions elsewhere in the plant would not be considered as spurious initiation of the safety system.

When considering the upgrading of the basic level based on the above factors, the following aspects require consideration:

(1) Some of the above factors may have already been included in the basic rating, e.g. common mode failure. It is therefore important to take care that such failures are not double counted. Allowing for all additional factors, the level of an event can only be upgraded by one level.
(2) The event should not be uprated beyond the maximum level derived in accordance with Section IV–2, and this maximum level should only be applied if, had one other event taken place (either an expected initiator or a further component failure), an accident would have occurred.

IV–3.3.1. Common cause failures

A common cause failure is the failure of a number of devices or components to perform their functions as a result of a single specific event or cause. In particular, it can cause the failure of redundant components or devices intended to perform the same safety function. This may imply that the reliability of the whole safety function could be much lower than expected. The severity of an event which implies a common cause failure affecting one or several components is therefore higher than a random failure affecting the same components.

Events where there is a difficulty in operating systems caused by missing or misleading information can also be considered for uprating on the basis of a common cause failure.

IV–3.3.2. Procedural inadequacies

The simultaneous challenge of several layers of defence in depth may arise because of inadequate procedures. Such inadequacies are therefore also a possible reason for uprating the level on the scale. Examples include: incorrect or inadequate instructions given to operators for coping with an event (during the Three Mile Island accident in 1979, the procedures to be used by operators in the case of safety injection actuation were not adapted for the particular situation of a loss of coolant in the steam phase of the pressurizer); or deficiencies in the surveillance programme highlighted by anomalies not discovered by normal procedures or plant unavailabilities well in excess of the test interval.

IV–3.3.3. Events with implications for safety culture

Safety culture has been defined as “that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority,

nuclear plant safety issues receive the attention warranted by their significance". A good safety culture helps to prevent incidents but, on the other hand, a lack of safety culture could result in operators performing in ways not in accordance with the assumptions of the design. Safety culture has therefore to be considered as part of the defence in depth and consequently, a deficiency in safety culture could justify upgrading the rating of an event by one level.

To merit upgrading due to a deficiency in the safety culture, the event has to be considered as a real indicator of a deficiency in the overall safety culture.

Examples of such indicators could be:

— A violation of operational limits and conditions or a violation of a procedure without justification (see Appendix V for additional information on OL&C and Technical Specifications);
— A deficiency in the quality assurance process;
— An accumulation of human errors;
— A failure to maintain proper control over radioactive materials, including releases into the environment or a failure in the systems of dose control;
— The repetition of an event, indicating that either the possible lessons have not been learnt or the corrective actions have not been taken after the first event.

It is important to note that the intention of this guidance is not to initiate a long and detailed assessment but to consider if there is an immediate judgement that can be made by those rating the event.

IV–4. DEFINITIONS

This section provides definitions for words not defined in other IAEA publications. In many cases a more detailed explanation is provided in this manual.

areas not expected by design. Areas whose design basis, for either permanent or temporary structures, does not assume that following an incident the area could receive and retain the level of contamination that has occurred and prevent the spread of contamination beyond the area. Examples of events involving contamination of areas not expected by design, are:

— Contamination by radionuclides outside controlled or supervised areas, where normally no activity is present like floors, staircases, auxiliary buildings, storage areas, etc.
— Contamination by plutonium or highly radioactive fission products of an area designed and equipped only for the handling of uranium.

authorized operating regime. See operating limits and conditions.

defence in depth. As defined in 'Basic Safety Principles for Nuclear Power Plants' (Safety Series No. 75-INSAG-3 Rev. 1) (see footnote 4):

> "To compensate for potential human and mechanical failures, a defence in depth concept is implemented, centred on several levels of protection including successive barriers preventing the release of radioactive material to the environment. The concept includes protection of the barriers by averting damage to the plant and to the barriers themselves. It includes further measures to protect the public and the environment from harm in case these barriers are not fully effective."

high integrity safety layer. Should have all of the following characteristics:

(a) The safety layer is designed to cope with all relevant design basis faults and is explicitly or implicitly recognized in the plant safety justification as requiring particularly high reliability or integrity.
(b) The integrity of the safety layer is assured through appropriate monitoring or inspection such that any degradation of integrity is identified.
(c) If any degradation of the layer is detected, there are clear means of coping with the event and of implementing corrective actions, either through pre-determined procedures or through long times being available to repair or mitigate the fault.

initiator (initiating event). An identified event that leads to a deviation from the normal operating state and challenges one or more safety functions.

operability of a safety function. The operability of a safety function can be 'Full', 'Within OL&C', 'Adequate' or 'Inadequate', depending upon the operability of the individual redundant and diverse safety systems and components.

operability of equipment. A component shall be considered operable when it is capable of performing its required function in the required manner.

operating area. Areas where worker access is permitted. It excludes areas where specific controls are required owing to the level of contamination or radiation.

operational limits and conditions (OL&C). A set of rules which set forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of the nuclear power plant (in most countries, these are included within 'Technical Specifications').

radiological barrier. A barrier designed to prevent dispersion of radioactive material beyond its intended containment.

radiological equivalence. The quantity of a radionuclide which must be released to give the same committed effective dose as the reference quantities of ^{131}I or

^{106}Ru under on-site and off-site impact, calculated using the model detailed in Appendix I.

safety functions. The three basic safety functions are: (a) controlling the reactivity or the process conditions; (b) cooling the radioactive material; and (c) confining the radioactive material.

safety layers. A safety provision that cannot be broken down into redundant parts.

safety provisions. Procedures, administrative controls, or passive or active systems which are usually provided in a redundant way, with their availability controlled by OL&C.

safety relevance. Concerns nuclear or radiological safety.

safety systems. Systems important to safety, provided to ensure the safety functions.

Part V

EXAMPLES TO ILLUSTRATE THE DEFENCE IN DEPTH RATING GUIDANCE

V–1. GUIDANCE ON THE USE OF THE LAYERS APPROACH FOR SPECIFIC TYPES OF EVENTS

V–1.1. Criticality control

The behaviour of a critical system and its radiological consequences are heavily dependent on the physical conditions and characteristics of the system. In homogeneous fissile solutions the possible number of fissions, the power level of the criticality excursion and the potential consequences of a criticality excursion are limited by these characteristics. Experience with criticality excursions in fissile solutions shows that typically the total number of fissions is in the order of 10^{17}–10^{18}.

Heterogeneous critical systems such as fuel rod lattices or dry solid critical systems have the potential for high power peaks leading to an explosive release of energy and the release of large amounts of radioactive material as a result of substantial damage to the installation.

The main hazard from a criticality excursion is due to high radiation fields from direct neutron and gamma radiation leading to potentially high radiation exposure to personnel. A second consequence might be off-site release of short lived radioactive fission products and potentially severe contamination within the facility. In addition, an explosive release of energy resulting from a criticality excursion in a heterogeneous system might also result in the release of the fissile material. Thus, in most cases off-site and on-site impact is limited to level 4. Only where fissile material can be released by an explosion is a higher rating possible.

In accordance with the general guidance:

— Minor deviations from the criticality safety regime which are within OL&C should be rated at level 0.
— Operation outside OL&C should be rated at least at level 1.

An event should be rated at level 3 if a criticality accident with maximum potential consequences of level 5 or higher would have occurred had conditions been less favourable or had one further failure in the safety provisions occurred. Level 2 would be appropriate for similar events if the potential could only have been level 3 or 4.

If more than one safety layer remains, then a lower rating would be appropriate as indicated in Table V.

V–1.2. Loss or removal of radioactive sources

This section considers events involving the loss or misplacement of sealed and unsealed radioactive sources whose storage and use are subject to administrative controls. Since such events result from the failure of the required control procedures, a minimum rating of level 1 is appropriate for all events involving the permanent loss of a source or the discovery of a source in an inappropriate location.

If the potential off-site consequences, should the source disintegrate, cannot reach those defined for level 5 but the source size is such that there is the potential for a person to receive a dose which would result in a fatal exposure or radiation burns (i.e. prompt adverse health effects), its permanent loss should be rated at level 2 under defence in depth. Similarly, the discovery of such a source outside the controlled area, or off-site, in a location that could eventually have led to adverse health effects, should also be rated at level 2.

If disintegration of the source could result in a level 5 event, its permanent loss should be rated at level 3 under defence in depth.

V–1.3. Unauthorized release/spread of contamination

Any event involving the transfer of contamination on-site or off-site which results in a level above the prescribed limit for the area may justify a rating of level 1 based on Section IV–3.3.3 (failure to maintain proper control over radioactive materials). More significant failures in safety provisions should be rated by considering the maximum potential consequences should all the safety provisions fail and the number of safety layers remaining.

If significant off-site contamination is not possible, then the maximum rating under defence in depth is level 2. Breaches of discharge authorizations should be rated at least at level 1.

V–1.4. Dose control

Occasionally, situations may arise when the radiological control procedures and managerial arrangements are inadequate and employees receive unplanned radiation exposures (internal and external). Such events may justify a rating of level 1 based on Section IV–3.3.3 (failure to maintain proper control over radioactive materials). If the event results in the cumulative dose exceeding prescribed limits, the event should be rated at least at level 1 as a violation of OL&C.

Level 2 would be appropriate under defence in depth if the maximum potential consequences should the safety provisions fail are level 3 or 4 and the event results in only one safety layer remaining. In general, the guidance in Section IV–3.3 should not be used to uprate events related to dose control failure from a basic rating of level 1. Otherwise events where dose was prevented will be rated at the same level as those where significant doses in excess of dose limits were actually incurred.

V–1.5. Interlocks on doors to shielded enclosures

Inadvertent entry to normally shielded locations is generally prevented by the use of radiation activated interlocking systems on the entrance doors, the use of entry authorization procedures and pre-entry checks on radiation dose rates.

Failure of the shield door interlocking protection can result from loss of electrical supply and/or defects in either the detector(s) or the associated electronic equipment.

As the maximum potential consequences for such events are limited to level 4, events where a further failure in the safety provisions would result in an accident should be rated at level 2. Events where additional safety layers remain, including administrative arrangements governing authorization for entry, should be rated at level 1.

V–1.6. Failures of extract ventilation, filtration and cleanup systems

Three separate but interrelated extract ventilation systems are often provided to maintain a pressure gradient between the plant vessels, cells/glove boxes and operating areas as well as adequate flow rates through apertures in the cell operating area boundary wall to prevent back diffusion of radioactive material. In addition, cleanup systems such as high efficiency particulate air (HEPA) filters or scrubbers are provided to reduce discharges to the atmosphere to below pre-defined limits and to prevent back diffusion into areas of lower activity.

The first step in rating events associated with the loss of such systems is to determine the maximum potential consequences both on-site and off-site should all the safety provisions fail. This should consider the material inventory and the possible means for its dispersion both inside and outside the plant. It is also necessary to consider the potential for a decrease in the concentration of inerting gases or the buildup of explosive mixtures. In most cases, unless an explosion is possible, it is unlikely that the maximum potential consequences would exceed level 3 and therefore the maximum under defence in depth would be level 2.

The second step is to identify the effectiveness of the remaining safety provisions, including procedures to prevent the generation of further activity by cessation of work. The rating of such events is illustrated by examples 16 and 17 in Section V–3.

V–1.7. Handling incidents and drops of heavy loads

V–1.7.1. Events not involving fuel assemblies

The impact of handling incidents or failure of lifting equipment depends on the material involved, the area in which the incident occurred and the equipment which was or could have been affected.

Events where a dropped load threatens a spillage of radioactive material (either from the dropped load itself or from affected pipework or vessels) should be rated by considering the maximum potential consequences and the likelihood that such a spillage might have occurred. Incidents where a dropped load causes only limited damage but has a relatively high probability of causing an accident should be rated at the maximum level under defence in depth. Similarly, events where only one safety layer remains and that layer is not considered to be of especially high reliability/integrity should also be rated at the maximum level.

Incidents where the likelihood is lower or there are additional safety layers should be rated following the guidance in Section IV–3.2.2. Minor handling incidents that would be expected over the lifetime of the plant should be rated at level 0.

V–1.7.2. Fuel handling faults

Events during the handling of unirradiated uranium fuel elements with no significant implications for the handling of irradiated fuel should typically be rated at level 0 if there has been no risk of damaging spent fuel elements or safety related equipment.

The radioactive inventory of a single fuel element is obviously much lower than the inventory of the spent fuel pool or the reactor core. As long as the cooling of the spent fuel element is guaranteed, this provides an important safety layer since the integrity of the fuel matrix is not affected by overheating. In general there will be very long time-scales associated with fuel overheating. Depending on the plant configuration, containment will also provide a safety layer in most cases.

Events expected over the lifetime of the plant which do not affect the cooling of the spent fuel element and only result in a minor release or no release typically should be classified at level 0.

Level 1 should be considered for events involving:

— Events not expected over the lifetime of the plant,
— Operation outside OL&C,
— Limited degradation of cooling not affecting the integrity of the fuel pins,
— Mechanical damage of fuel pin integrity without degradation of cooling.

Level 2 may be appropriate for events in which there is damage to the fuel pin integrity as a result of substantial heat up of the fuel element.

V–1.8. Loss of electrical power supply

At many plants it is often necessary to provide a guaranteed electricity supply to ensure continued safe operation and to maintain the availability of monitoring equipment and surveillance instruments. Several independent electrical supply routes and diverse supply means are used to prevent common cause failure. While most plants will be automatically shut down to a safe condition on total loss of electrical power supplies, in some plants additional safety provisions, such as the use of inerting gas, will be provided.

For some plants there will be no adverse safety effects even with a complete loss of power supply lasting several days; such events at these facilities should generally be rated at level 0 or 1 as there should be several means available to restore the power supply within the available time. Level 1 would be appropriate if the availability of safety systems had been outside OL&C.

In order to rate events involving loss of off-site supplies or failures in on-site supply systems, it is necessary to use the general guidance in Section IV–3.2.2, taking account of the extent of any remaining supplies, the time during which the supplies were unavailable and the maximum potential consequences. It is particularly important to take account of the time delay acceptable before restoration of supplies is required.

Partial loss of electrical power or loss of electrical power from the normal grid with available power supply from stand-by systems is expected over the life of the plant and therefore should be rated below scale.

V–1.9. Fire and explosion

A fire or explosion within or adjacent to the plant which does not have the potential to degrade any safety provisions should be rated level 0 or out of scale. Fires which are extinguished by the installed protection systems, functioning as intended by design, should also be rated level 0 or out of scale.

The significance of fires and explosions at nuclear installations depends not only on the material involved, but also on the location and the ease with which fire fighting operations can be undertaken. The rating depends on the maximum potential off-site or on-site consequences, the number and effectiveness of the remaining safety layers, including barriers and safety systems. The effectiveness of the remaining safety layers should take account of the likelihood that they could have been degraded. Any fire or explosion involving low level waste should be rated at level 1 owing to deficiencies in procedures or safety culture.

V–1.10. External hazards

The occurrence of hazards such as earthquakes, tornadoes or explosions may be rated in the same way as other events, by considering the effectiveness of the remaining safety provisions. For events involving failures in systems specifically provided for protection against hazards, the number of safety layers should be assessed, including the likelihood of the hazard occurring during the time when the system was unavailable. Owing to the low expected frequency of such hazards, a rating greater than level 1 is unlikely to be appropriate.

V–1.11. Events during transport

As with many events, it is very important to establish the maximum potential consequences and hence the maximum rating under defence in depth. The transport regulations control the maximum activity which can be contained within each package, consignment or vehicle. This maximum activity may be related to the parameter A_2, where A_2 is the maximum nuclide specific radioactive contents allowed in a Type A package when the material is in other than special form. It is therefore possible to relate the transported activity in terms of the applicable A_2 value to the maximum possible consequences on INES by assuming 100% release of the contents, and to the maximum under defence in depth. Table VI shows the relationship between transported activity and consequences which should be used for guidance on rating transport events involving airborne releases.[7]

On the basis of the above and the general principles for rating events using the safety layers approach, the following specific guidance (given in Table VII) can be derived for particular cases. For other cases, the rating will need to take account of the adequacy of the remaining safety provisions using the general guidance.

V–1.12. Failures in cooling systems

V–1.12.1. Events during reactor shutdown

Most reactor safety systems have been designed to cope with initiators occurring during power operation. Events in hot shutdown or startup conditions are quite similar to events in power operation and should be treated as discussed in Section IV–3.2.1. Once the reactor is shut down, some of these safety systems are still required to assure

[7] The INES guidance on radiological equivalence is for airborne releases only. It is not possible to provide generic guidance on equivalence for aquatic releases.

TABLE VI. RELATIONSHIP BETWEEN TRANSPORTED ACTIVITY AND MAXIMUM RATING

Transported activity	Maximum potential consequences (based on assumption of 100% release of the contents)	Maximum rating under defence in depth
Greater than 100 A_2	Level 5–7	3
A_2 to 100 A_2	Level 3–4	2
Less than A_2	Level 2	1

TABLE VII. RATING OF TRANSPORT EVENTS

Reduction of safety layers	Transported activity of package		
	< A_2	A_2 – 100 A_2	> 100 A_2
Events not involving a transportation accident			
Only one safety provision remaining[a]	0	1	2
No safety provisions remaining (e.g. inadequate package)	1	2	3
Loss of package	1	2	3
Events involving a transportation accident			
No degradation of safety provisions	0	0	0
Major degradation of safety[a] provisions (only one or no safety provision remaining)	1	2	3

[a] Unless the provision meets the requirements of a high integrity layer.

the safety functions, but usually more time is available before a possible release of the core inventory can occur.

On the other hand, the time available for manual actions to prevent a major increase in fuel temperature and a release of radioactive fission products may replace part of the safety provisions in terms of redundancy or diversity, i.e. depending on the status of the plant a reduction in the redundancy of safety equipment and/or barriers may be acceptable during some periods of cold shutdown. In such shutdown conditions, the configurations of the barriers are sometimes also quite different (for example, open primary coolant system and open containment).

Some examples applicable to pressurized water reactors are presented in Section V–2 to give guidance for rating events during cold shutdown following the safety layers approach. The rating mainly takes into account the time available for corrective actions and the number of safety layers not affected. For other reactor types it will be necessary to use this as illustrative guidance together with the general principles to rate such events.

V–1.12.2. Events affecting the spent fuel pool

After some years of operation, the radioactive inventory of the spent fuel pool may be high. In this case, rating of events affecting the spent fuel pool with respect to the impact on defence in depth may span the full range from below scale up to level 3.

Because of the large water inventory and the comparably low decay heat, there is usually plenty of time available for corrective actions to be taken for events involving degradation of spent fuel pool cooling. This is equally true for a loss of coolant from the spent fuel pool, since the leakage from the pool is limited by design. Thus, a failure of the spent fuel pool cooling system for some hours or a coolant leakage will not usually affect the spent fuel. Therefore, minor degradation of the pool cooling system or minor leakages should typically be rated at level 0.

Operation outside OL&C or a substantial increase in temperature or decrease of the spent fuel pool coolant level should be rated at level 1. An indication of level 2 could be the start of fuel element uncovering. Substantial fuel element uncovering and heatup clearly indicate level 3.

V–1.12.3. Other installations

Failures in essential cooling systems can be rated in a similar way to failures in electrical systems by taking account of the maximum potential consequences, the number of safety layers remaining and the time delay acceptable before restoration of cooling is required.

In the case of failures in the cooling systems of high level liquid waste or plutonium storage, level 3 is likely to be appropriate for events where only a single safety layer remains for a significant period of time.

V–2. ILLUSTRATIVE EXAMPLES OF APPLYING THE SAFETY LAYERS APPROACH

To illustrate the use of the guidance in Section IV–3.2.2, a number of examples based on cooling a shut down reactor are discussed below.

Example 1

Event description

In this first example, shutdown cooling is provided by circulation of coolant through a residual heat removal (RHR) heat exchanger via a single suction pipe with two isolation valves. The primary circuit is closed. In the event of closure of the isolating valves, the coolant temperature will rise but will take approximately one hour to reach unacceptable temperatures. The valves are operable from the control room. Steam generators are open for work and are therefore unavailable. Safety injection is not available, HPSI pumps are separate from the charging pumps and relief valves are available to control primary circuit pressure. The event to be rated is one where spurious operation of pressure sensors caused the isolation valves to close. Alarms in the control room notified the operator of the valve closure and, having checked that the pressure rise was a spurious signal, the valves were reopened. Temperatures did not rise above OL&C.

Rating explanation

The maximum potential consequences from loss of cooling exceed level 4 and therefore the maximum rating under defence in depth is level 3. The safety function of concern is cooling of the fuel. Ultimately, the only safety layer that provides cooling is cooling of the primary coolant through the single RHR suction pipe, i.e. there is only one safety layer.

It is therefore necessary to consider the integrity of that single safety layer, and to consider both the hardware and software aspects. Considering firstly the operator actions required, in order to restore cooling the operator must ensure that the pressure signal was spurious and, if the rise in coolant temperature has caused a subsequent rise in pressure, the pressure needs to be reduced. A procedure for reinstating RHR after closure of the valves does exist. The operation can be carried out in the time available but not with a large margin. Considering the hardware aspects, the failure of either valve to reopen will result in the unavailability of the safety layer. Also, there is certainly not sufficient time to carry out any repairs should the valves fail to open.

For these reasons it is not considered that the single layer is a high integrity safety layer, even though it was the only layer provided by design. The need to be able to open both of the isolating valves in order to reinstate supplies clearly limits the integrity of the safety layer. Such an event at a plant of the design described would therefore be level 3.

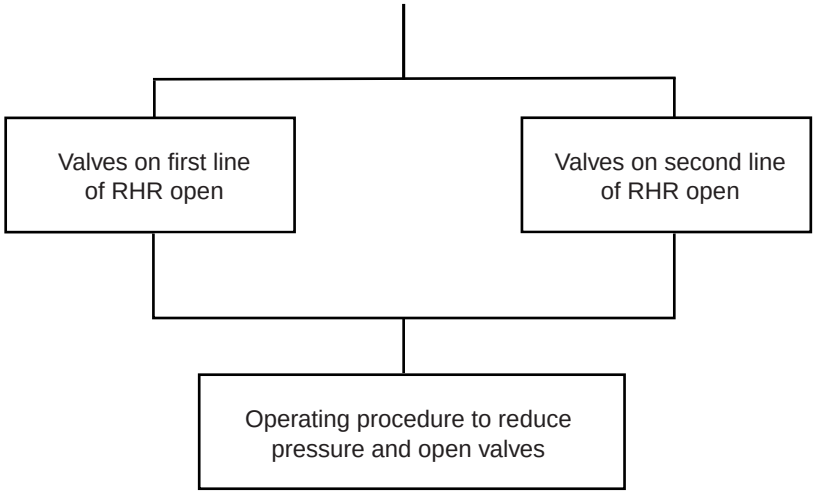
Example 2

Event description

In this example, the design is modified slightly from example 1. Now there are two separate RHR lines, each with two isolating valves, with the valves in each line fed by separate pressure transducers. The event is similar, except that the pressure increase is genuine.

Rating explanation

There now appear to be two layers as far as hardware is concerned. However, both still rely on the operator to reopen the valves. The safety provisions can be illustrated as follows:



The reliability of the safety provisions is limited by the need for operator action. Given the complexity of the operation and the limited time available, it is considered that there is only one effective safety layer, i.e. an operating procedure requiring pressure reduction and reopening of the isolation valve. Again, therefore, level 3 is considered appropriate.

Example 3

Event description

The design for this example is the same as for example 2. However, the event is assumed to occur some time after the reactor has been shut down. It is assumed that there are five hours to carry out the required actions.

Rating explanation

As before there are two hardware safety layers, and a software safety layer in series, but there is now a significantly longer period of time to carry out the required actions. The available operator action can therefore be regarded as a high integrity safety layer. The limiting aspect of the safety provisions is now the two hardware layers. The existence of two hardware layers means that the event should be rated at level 2.

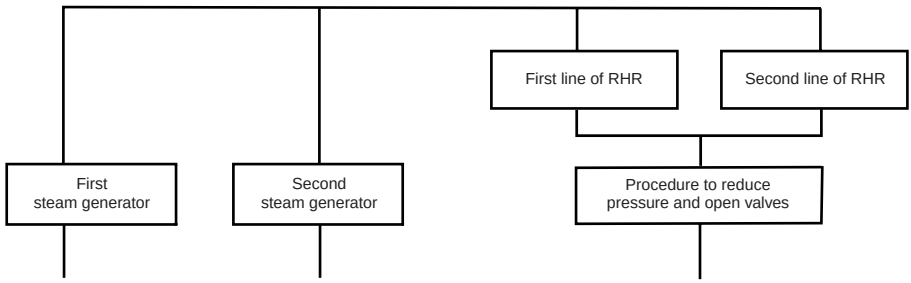
Example 4

Event description

In this example, the design is the same as for example 2, but two steam generators are also available. The event to be rated is also the same as for example 2.

Rating explanation

There are now four hardware layers, but in addition the availability of steam generators provides a much longer time-scale for the required operator actions and allows time for repairs to be carried out. The safety provisions are illustrated below. As a result of the longer time-scales available, all four layers can be considered as fully effective and a rating of zero is considered appropriate:



Example 5

Event description

This example is based on the design of example 1, but one week after shutdown, when the cavity is open and flooded. Loss of RHR will now only result in a very slow heatup of the primary coolant, allowing some ten hours for operator action.

Rating explanation

Considering the safety function of fuel cooling, there are now two safety layers. The first is the RHR system and the second is the ability to add water so as to maintain the water level as water and heat are lost through evaporation. The second layer can be considered as a high integrity layer for the following reasons:

— There are long times available for the operator to take action;
— There are a number of ways of adding more water (e.g. LSSI, fire hoses, etc.), though the boron concentration must be controlled;
— This safety layer is recognized in the safety justification as a key safety feature.

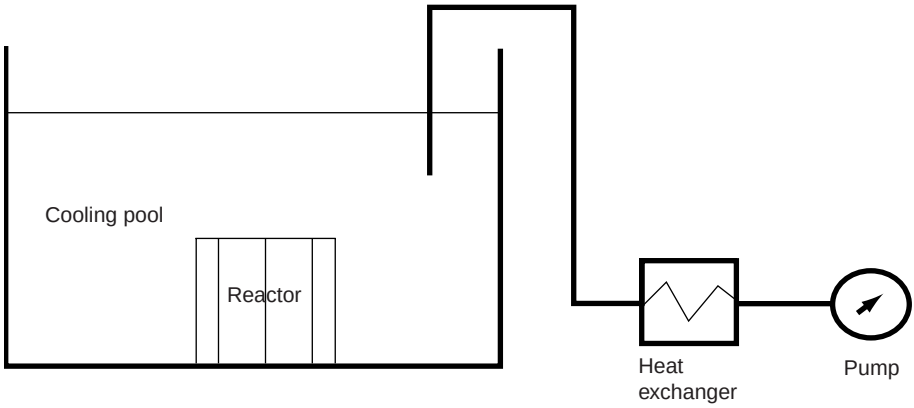
In addition, the time available is such that the first layer is of greater integrity than assumed in example 1 as there is adequate time for repair. In this case looking at the temperature transient provides a way of measuring the time elapsed and the time available. The following guidance is therefore applicable:

— Minor changes in the coolant temperature should typically be rated below scale,
— Exceeding maximum allowed coolant temperatures or coolant temperature differences (versus time) specified in OL&C should be rated at level 1,
— Substantial heatup of the coolant, e.g. massive boiling, should be rated at level 2,
— Start of significant uncovering of fuel elements would typically indicate level 3.

Example 6

Event description

This example is based on a 100 kW research reactor with a large cooling pool and a heat exchanger/purification system, as shown below. In the event of loss of cooling any heatup of the water will be extremely slow.



The event to be rated is one where the pipework downstream of the pump failed and coolant was pumped out to the bottom of the suction pipe. The pump then failed owing to cavitation.

Rating explanation

There are two safety functions to be considered, one is the cooling of the fuel and the other is the shielding to prevent high worker doses. It is necessary to consider initially the maximum potential consequences should all safety layers fail. For both safety functions, owing to the low inventory, the maximum potential consequences cannot exceed level 4 and therefore the maximum under defence in depth is level 2. Considering the cooling function by design there are three safety layers: one is the heat exchanger system, another is the large volume of water in the pool and the third is the ability to cool the fuel in air. The suction side has been deliberately designed so as to ensure that a large volume of water remains in the pool should the pipe fail. Furthermore, it is clear that the main safety layer is the volume of water. This can therefore be considered as a high integrity layer for the following reasons:

— The heat input is small compared with the volume of the water such that any heatup will be extremely slow. It should take many days for the water level to decrease significantly.
— Any reduction in water level would be readily detected by the operator and the water level could be simply topped up via a number of routes.
— The safety justification for the plant recognizes this as the key safety layer, and demonstrates its integrity. The suction pipe to the heat exchanger was carefully designed to ensure that adequate water remained.

As such the basic rating is considered to be zero as there are two safety layers remaining and one is of high integrity. Considering the shielding safety function, there is only one safety layer remaining, but as it is of high integrity, the basic rating is considered to be zero.

V–3. WORKED EXAMPLES BASED ON REAL EVENTS

V–3.1. Examples using the initiator approach

Example 1: Reactor scram following the fall of control rods — level 0

Event description

The unit was operating at rated power. During insertion of the shutdown rod (bank A), which was carried out under the periodic control rod surveillance test, the reactor was scrammed as a result of the 'power range neutron flux high negative rate' signal, which also caused automatic turbine and generator trip. The control rod operation was promptly checked on the control rod transient position detector. It was found that the four control rods of shutdown bank A group had fallen prior to reactor shutdown.

An inspection of the control circuit of the control rod drive mechanism showed that the cause of the malfunction was a defective regulation card (printed circuit board). Later, the relevant faulty card was replaced with a spare card, and after the integrity of the control circuit had been checked the operation was resumed at rated output.

Rating explanation

Off-site and on-site impact is not relevant for rating. The accidental insertion of control rods does not challenge the safety functions and is therefore not an initiator. The reactor trip is an initiator (expected), and the safety function 'cooling of the fuel' was fully available. Following Section IV–3.2.1.3(a), box A1 of the table is appropriate. There are no reasons for uprating and therefore level 0 is selected.

Example 2: Reactor coolant leak during on-power refuelling — level 1

Event description

During routine refuelling at full power, a reactor coolant leak of 1.4 t/h developed in the fuelling vault. Operators determined that the east fuelling bridge had

dropped 40 cm. The reactor was shut down and cooled. Coolant pressure was maintained by transfer from other units and recovery from sump. Total leakage was 22 t (~10% of the inventory). No safety system operation was required with the exception of containment box-up on high activity after one hour. There was no abnormal release of radioactivity to the environment.

Rating explanation

Off-site and on-site impact is not relevant for rating.

Although there was a very small reactor coolant leak, there was no challenge to the safety functions, as operator action maintained water inventory. Had the leak developed into a small loss of coolant accident (LOCA), all the required safety systems would have been fully available. Thus level 0 is appropriate.

The cause of the problem was interlock failure which was not checked by the surveillance programme. Also, this deficiency was known before the event. For these reasons, the event was uprated to level 1 (see Section IV–3.3).

Example 3: Containment spray not available because valves left in closed position — level 1

Event description

The twin plant has to shut down both its reactors annually in order to perform the required tests on the common emergency core cooling system (ECCS) and the related automatic safety actions. These tests are usually performed when one of the two reactors is in cold shutdown for refuelling.

On 9 October, units 1 and 2 were subjected to these tests. Unit 1 remained in the cold shutdown condition for refuelling and unit 2 resumed power operation on 14 October. On 1 November, it was discovered during the monthly check of the safeguard valves that the four valves on the discharge side of the containment spray pumps were closed. It was concluded that these valves had not been reopened after the tests on 9 October, in contradiction to the requirements of the related test procedure. Unit 2 had thus operated for 18 days with spray unavailable.

It was concluded that the cause of the event was a human error. However, it was recognized that the error occurred at the end of a test period that was longer than usual (as a result of troubleshooting) and that a more formal reporting of actions accomplished could have been very useful.

Rating explanation

Off-site and on-site impact is not relevant for rating.

There was no real initiator, the operability of the safety function 'confinement' was degraded. The operability was less than the 'minimum required by OL&C', but more than just adequate as a diverse system was available. The initiator that would challenge the degraded safety function was a large LOCA (unlikely). Following Section IV–3.2.1.3(b), box C3 of the table is appropriate. The fault was caused by human error but it is not considered appropriate to uprate the event owing to a deficiency in safety culture. (Appendix III explains that the choice of level 1 rather than zero for the basic rating already took account of the fact that OL&C had been violated.)

Example 4: Primary system water leak through the rupture disc of the pressurizer discharge tank — level 1

Event description

The unit had been brought to hot shutdown. The RHR system had been isolated and partially drained for system tests after modification work and was therefore not available. The periodic test of pressurizer spray system efficiency was under way and the reactor coolant system was at a pressure of 159 bar. At about 16:00, the pressurizer relief tank high pressure alarm was actuated. The level in the volume control tank fell, indicating leakage of reactor coolant at an estimated rate of 1.5 m^3 an hour. The operator went into the reactor building in an attempt to discover where the leak was located and concluded that it was coming from the stem of a valve on the reactor coolant system (manual valve located on the temperature sensor bypass line). The operator checked that the valve was leaktight by placing it in its back seat position by means of the handwheel (in fact, the valve was still not correctly seated). The leakage continued and maintenance staff were called in at 18:00, but they too failed to find the source of the leak.

During this time, the pressure and temperature inside the pressurizer relief tank continued to rise. The operator maintained the temperature below 50°C by means of feed and bleed operations, i.e. injections of cold make-up water and drainage into the reactor coolant drain recovery tank. Two pumps installed in parallel directed this effluent out of the reactor building towards the boron recycle system tank.

At around 21:00, the activity sensors indicated an increase in radioactivity in the reactor building. At 21:56, the set point for partial isolation of the containment was reached. This resulted notably in closure of the valves inside the containment on the nuclear island vent and drain system. At this point effluent could no longer be routed to the boron recycle system. The pressure inside the pressure relief tank continued to rise until at 21:22 the rupture discs blew. To maintain the temperature in the pressurizer relief tank at around 50°C, water make-up had to be continued until 23:36. At 01:45, activity levels inside the reactor building fell below the set point for containment isolation.

At 02:32, the reactor coolant system was at a pressure of 25 bar; the unit had been brought to subcritical hot shutdown condition with heat being removed by the steam generators; the RHR system was still unavailable.

The RHR system was reinstated at 10:54 and at 11:45 the leaking valve on the reactor coolant system was disconnected from its remote control to allow it to be reseated, thereby stopping the leak.

Rating explanation

Off-site and on-site impact is not relevant for rating. No real initiator occurred as the emergency core cooling safety systems were not challenged. The initial leakage was controlled by the normal make-up systems (see Section IV–3.2.1.1). Level 0 is therefore appropriate.

The spurious initiator of containment isolation caused operating difficulties and gave misleading information. For these reasons, the event was uprated to level 1 (see Section IV–3.3).

Example 5: Loss of forced gas circulation for between 15 and 20 minutes — level 2

Event description

A single phase fault on the instrument supplies to reactor 1 was not cleared automatically and persisted until supplies were changed over manually. The fault caused both high pressure and low pressure feed trip valves to close on one boiler, leading to rundown of the corresponding steam driven gas circulator. Much of the instrumentation and automatic control on the boilers and reactor 1 was lost. Manual rod insertion was possible and was attempted, but the rate was insufficient to prevent rising temperatures, leading to reactor 1 being automatically tripped on high absolute fuel element temperature (approximately 16°C rise). It appeared to the operator that all the rod control systems were rendered inoperable. The battery backed essential instrumentation and the reactor protection system remained functional, together with some of the normal control and instrumentation systems.

All gas circulators ran down as the steam to their turbines deteriorated. The instrument supplies fault prevented engagement of gas circulator pony motors either automatically or manually. Low pressure feed was maintained throughout to three out of four boilers and was restored to the fourth boiler by operator action. After the initial transient, leading to reactor tripping, fuel element temperatures fell but rose as forced gas circulation failed. These temperatures stabilized at about 50°C below normal operational levels before falling once again when gas circulator pony motors were started on engagement of stand-by instrument supplies. Reactor 2 was unaffected and operated at full output throughout. Reactor 1 was returned to power the following day.

Rating explanation

Off-site and on-site impact is not relevant for rating. This event needs to be considered in two parts. The first initiator was the transient caused by loss of feed to one boiler together with loss of indications. This challenged the protection system, which was still fully available. This part of the event would therefore be rated at level 0. It should be noted that although the first occurrence in the event was a fault in the instrument supplies, this is not the initiator. The instrument fault caused feed to be lost to one boiler but did not directly challenge any safety systems. It is not therefore to be considered an initiator. The transient that followed challenged the protection system and is therefore an initiator.

The second initiator was the reactor trip and rundown of the steam driven gas circulators. This challenged the safety function 'cooling of the fuel'. The operability of this safety function was less than the 'minimum required by OL&C' as none of the pony motors could be started, but more than adequate as natural circulation provided effective cooling and forced circulation was restored before temperatures could have risen to unacceptable levels. Following Section IV–3.2.1.3(a), box C1 of the table is appropriate, giving a rating of 2 or 3. As explained in that section, the level chosen depends on the extent to which the operability is greater than just adequate. In this event because of the availability of natural circulation and the limited time for which forced circulation was unavailable, level 2 is appropriate.

Regarding possible uprating, there are two issues to be considered, both identified in Section IV–3.3. The fault involved common mode failure of all the circulators. However, this fact has already been taken into account in the basic rating and to uprate the event would be double counting (see the introduction to Section IV–3.3, item (a)). The other relevant factor is the difficulty caused by absent indications. However, this was more relevant to controlling the initial transient and could not have led to a worsening of the post-trip cooling situation. Furthermore, from item (c) of Section IV–3.3, level 3 would be inappropriate, as a single further component failure would not have led to an accident.

Example 6: Fuel assembly drop during refuelling — level 1

Event description

While performing refuelling after lifting the fuel assembly from its cell, spontaneous pull-out of the refuelling machine telescopic beam occurred and a fresh fuel assembly slumped onto the central tube of the refuelling machine flask. Interlocks operated as designed and no fuel damage or depressurization occurred.

Rating explanation

Off-site and on-site impact is not relevant for rating. Although the event only involved unirradiated fuel, it could have occurred with irradiated fuel. This needs to be taken into account in rating the implications for defence in depth. Dropping a single fuel assembly is identified as a possible initiator in Appendix IV, and following Section IV–3.2.1.3 a rating of 1 is appropriate as the provided safety systems were fully available (box A2 of the table). Application of the guidance in Section V–1.7.2 would give the same rating. There are no reasons for uprating the event.

Example 7: Partial blockage of the water intake of one unit and loss of off-site power at the twin unit during cold weather — level 3

Event description

There were two events, both having the same cause: partial blockage of unit 1 water intake and, two hours later, loss of off-site power at unit 2. In order to simplify the example, the impact on unit 2 only is considered here. The source of the twofold incident was the cold weather prevailing in the area at the time: ice floes blocked the water intake while the low temperatures contributed to the tripping of the conventional unit, followed by a voltage reduction on the transmission grid.

Blocking of the pumping station at unit 1 could have occurred as follows. Ice probably slipped under the skimmer, reaching the trash racks of the unit 1 pumping station. Further ice formation may have turned the ice floes into a solid block, partially obstructing the trash racks shared by the two screening drums of the unit 1 pumping station. This would have produced a significant reduction in raw water intake at the pumping station. There was no clear alarm signal indicating the drop in level.

As a result of the drop in level, vacuum loss at the condensers led to automatic tripping of the four auxiliary turbine generator sets at the site (between 09:30 and 09:34); the four corresponding busbars were each resupplied from the grid within one second.

The main turbine generator sets for unit 1 were switched off at 09:28 and 09:34 and the reactor was shut down.

Unit 2 remained in operation, although from 09:33 to 10:35 no auxiliary turbine generator set at the site was available (this situation was foreseen under general operating rules) and the only power supplies consisted of the transmission grid and the two main turbine generator sets for the unit. From 10:55 onwards, when a second auxiliary turbine generator was reconnected to its switchboard, two turboblowers were fed by the auxiliary turbine generators in operation and the two other turboblowers drawing from one of the two 400 kV lines.

At 11:43, following voltage reduction in the transmission grid, the two main turbine generator sets at unit 2 tripped almost simultaneously (unsuccessful house load operation), causing rod drop and reactor scram as well as loss of off-site power (tripping of line circuit breakers).

At this time, only two of the four auxiliary turbine generators had been brought back into service. Consequently, only two of the four turboblowers remained in operation to provide core cooling. The power lines linking unit 2 to the grid were restored after 10 and 26 minutes, so that the other turboblowers were brought back into service.

Rating explanation

Off-site and on-site impact is not relevant for rating. This is a complex set of events, but the event being rated is the operation of unit 2 without any on-site essential electrical supplies (owing to the loss of cooling water following ice formation). There was no initiator but the safety function 'cooling of the fuel' was degraded. The operability of the safety function was inadequate as there were no on-site electrical supplies to cope with a loss of off-site power (an expected initiator). Following Section IV–3.2.1.3(b), box D1 of the table is appropriate, giving a rating of level 3. Although the time of unavailability was short (one hour), the likelihood of loss of off-site power was high. Indeed, it was lost shortly afterwards. It is not appropriate, therefore, to downrate the event.

Example 8: Incorrect calibration of regional overpower detectors — level 1

Event description

During a routine calibration of the regional overpower detectors for shutdown systems 1 and 2, an incorrect calibration factor was applied. The calibration factor used was for 96% power, though the reactor was at 100% power. This error in calibration was discovered approximately six hours later, at which time all detectors were recalibrated to the correct value for operation at full power. The trip effectiveness of this parameter for both shutdown systems was therefore reduced for approximately six hours.

Rating explanation

Off-site and on-site impact is not relevant for rating. There was no real initiator but the operability of the protection system was reduced. The operability was less than the 'minimum allowed by OL&C', but greater than just adequate, as a second trip parameter with redundancy remained available. The wrongly calibrated

detectors would also have provided protection for most fault conditions. The protection was required for 'expected' initiators. Following Section IV–3.2.1.3(b), box C1 of the table is appropriate, giving level 1 or 2. Level 1 was chosen as the operability was considerably more than just adequate.

In considering whether the basic rating should be adjusted, it is relevant to consider that the fault only existed for a short time. On the other hand, there were deficiencies in the procedure. It was decided to keep the rating at level 1.

Example 9: Failure of safety system train during routine testing — level 1

Event description

The unit was operating at nominal power. During the routine testing of one diesel generator, a failure of the diesel generator control system occurred. The diesel was taken out of service for about six hours for maintenance and then returned to service. The Technical Specifications require that if one diesel generator is taken out of service, the other two safety system trains should be tested. This testing was not carried out at the time. Subsequently, the other safety system trains were tested and shown to be available.

Rating explanation

The explanation given here is appropriate for rating the event once the additional testing had been carried out to show that two trains were in fact available.

Off-site and on-site impact is not relevant for rating. There was no initiator but the safety function 'cooling of the fuel' was degraded. The operability was not less than the 'minimum allowed by OL&C', as two trains remained available. Following Section IV–3.2.1.3(b), box A1 of the table is appropriate, giving a basic rating of zero. However, the operators violated the Technical Specifications and in accordance with the guidance in Section IV–3.3 the event was uprated to level 1.

Example 10: Small primary circuit leak — level 2

Event description

A very small leak (detected only by humidity measurement) was discovered in the non-isolatable part of one safety injection line owing to defects which were not expected by the surveillance programme (the area was not inspected by the surveillance programme). Similar but smaller defects were present in the other safety injection lines.

Rating explanation

Following Section IV–3.2.3, if the defect had led to failure of the component, a large LOCA (an unlikely initiator) would have occurred. Using Section IV–3.2.1.3(a), box A3 of the table gives an upper value to the basic rating of 2. As only a leak occurred (with no actual failure of the pipework) the rating should be reduced by one level. However, as the defects could have led to common mode failure of all safety injection lines, the rating was upgraded to level 2.

Example 11: Unit scram caused by grid disturbances due to a tornado — level 3

Event description

The unit was operating stably at its rated power. As a result of a tornado, transmission lines were damaged. The unit was tripped by system emergency protection owing to strong frequency oscillations in the system.

Unit auxiliary power was supplied from the service transformer. Main steam header pressure was maintained and residual heat removed. Core cooling was maintained through natural circulation.

On voltage decrease, the diesel start signal was formed but diesel generators (DGs) failed to get connected to essential buses. Since the signal for DG start persisted, periodic restarts followed. Subsequent attempts to supply power to auxiliary buses from DGs were unsuccessful due to absence of air in the start-up bottles.

Four hours after the trip, total loss of power occurred. Half an hour later, unit power supply from the off-site source was restored. Throughout the transient, the core status was being monitored with the help of design-provided instrumentation.

Rating explanation

Off-site and on-site impact is not relevant for rating. The event was rated under ‘Impact on defence in depth’. A real initiator occurred, with loss of off-site AC power sources, including voltage and frequency fluctuations, due to a tornado. The frequency of this initiator is expected. The availability of the safety function was just adequate owing to the limited time of loss of off-site supplies.

According to Section IV.3.2.1.3(a), level 2 or 3 is assigned. As the safety function was only just adequate, level 3 was chosen. In addition, violation of OL&C occurred as efforts to bring the reactor to the minimum controlled power level were initiated with no DGs available to perform the safety function at unit total loss of power.

Example 12: Complete station blackout owing to a fire in the turbine building — level 3

Event description

When a PHWR was at power, a fire occurred in the turbine building. The reactor was tripped manually and a cooldown of the reactor was initiated.

Owing to the fire, many cables and other electrical equipment were damaged which resulted in a complete station blackout. Core decay heat removal was through natural circulation. Water was fed to the secondary side of the steam generators using diesel fire pumps. Borated heavy water was added to the moderator to maintain the reactor in a subcritical state at all stages.

Rating explanation

The event had neither off-site nor on-site impact. Loss of on-site electrical power (class IV, III, II or I) is a possible initiator for PHWR reactors which actually occurred (i.e. real). The safety function 'cooling' was adequate because the secondary side was fed using a diesel fire pump, which is not a normal safety system. According to Section IV–3.2.1.3(a), the event was rated at levels 2/3. Level 3 was chosen because of common cause failures (fire and degradation of the available safety systems owing to the loss of many indications) such that a number of potential further single failures could have resulted in an accident.

V–3.2. Examples based on the layers approach

Example 13: Pressurization of a fuel element dissolver vessel ullage — level 0

Event description

The detection of a small pressurization of the ullage space in a reprocessing plant dissolver resulted in the automatic shutting down of the process. The dissolver heating system was switched off and cooling water applied; the nitric acid feed to the vessel was stopped and the dissolution reaction suppressed by the addition of water to the vessel contents. No release of airborne contamination to the plant operating area or the environment occurred. Subsequent investigations indicated that the pressurization was due to an abnormal release of vapour and an increased rate of nitrous vapour production as a result of a short term enhanced rate of dissolution of the fuel.

Rating explanation

The event had neither off-site nor on-site impact. Because of the deviation in the process conditions, the process was automatically shut down; all steps of shut-down proceeded normally. No safety layers failed. Therefore, the basic rating of level 0 was selected and there are no reasons to uprate the event.

Example 14: Worker received a cumulative whole body dose above the dose limit — level 1

Event description

The whole body dose received by a plant manager during the last two weeks in December was marginally higher than authorized or expected, and as a result his cumulative whole body dose exceeded the annual dose limit.

Rating explanation

The event had no off-site impact and the on-site impact was below the threshold of significance. The basic rating is level 0 as there was no degradation of the safety layers provided to prevent significant doses to workers. However, since the annual limit of the cumulative whole body dose was exceeded, the event should be rated at level 1 according to Section IV–3.3.

Example 15: Failure of shield door interlocking system — level 2

Event description

The incident occurred when a container of highly radioactive vitrified waste was moved into a cell while the shield doors to the cell were open following a maintenance operation. The opening of the doors was controlled by a key exchange system, installed gamma interlocks and programmable logic controllers. The original design of the cell access system was modified twice during the commissioning period in an attempt to improve it. All of these systems failed to prevent the transfer of highly radioactive material into the cell while the shield doors were open.

Entry of personnel to this area is controlled by a permit which requires the wearing of personal alarm dosimeters. Personnel who might have been present in the cell or adjacent areas could have received a serious radiation exposure if they had failed to respond either to the container movement or to their personal alarm dosimeter

sounding a warning. In the event, the operator quickly observed the problem and closed the shield doors and no one received any additional exposure.

The plant design concerning access to the cells had been modified during commissioning and the consequences of these changes had been inadequately considered. In particular:

(a) The commissioning of the interlock key exchange system for the cell shield doors had failed to show that the system was inadequate.
(b) A programmable logic control system had not been programmed and commissioned correctly.
(c) The modifications were poorly assessed and controlled because their safety significance was not classified correctly.
(d) Designers and commissioning staff did not communicate properly.

A permit to work authorization had been closed, indicating that the plant had been returned to its normal state, but in fact it had not. The Temporary Plant Modification Proposal (TPMP) system was too frequently used in this plant and inadequately controlled, and the full TPMP system in use required improvement. In addition, the training and supervision of active cell entries were inadequate.

Rating explanation

Despite the failure of a number of safety layers, there was a remaining safety layer, namely the permit to work authorization procedure for entry to the cells requiring the use of personal alarm dosimeters. The maximum potential consequences for such activities is level 4 (death of a worker) and hence the basic rating of level 2 is appropriate.

Example 16: Failure of criticality control — level 1

Event description

A routine check of compliance with the operating rules in a fuel fabrication plant showed that six samples of fuel pellets had been incorrectly packaged. In addition to the permitted packaging, each sample had been placed in a plastic container. The additional plastic container contained the requirement that “no hydrogenous material in addition to the permitted wrapping” had to be introduced to the store. Subsequent investigation showed that the criticality clearance certificate was difficult to interpret and the related criticality assessment was inadequate to allow full understanding of the safety assessment.

Rating explanation

The maximum potential consequences of a criticality would be level 4, i.e. death of a worker. The maximum rating under defence in depth would therefore be level 2 (Section IV–3.2.2.3). The remaining safety layers are:

— Controls in place to prevent flooding (assumed in the safety case),
— Inspections to detect deviations from assumptions made in the safety case (e.g. the presence of other hydrogenous material).

There are therefore two safety layers remaining and the basic rating is level 1. This level would also be appropriate because:

— The operations were outside OL&C.
— The failure of safety culture to ensure adequate assessments and documentation.

Example 17: Prolonged loss of ventilation at a fuel fabrication facility — level 1

Event description

Following a loss of normal and emergency ventilation and non-compliance with procedures, the operators worked for over an hour without dynamic containment. The ventilation performs a dual role. Firstly, it directs radioactivity likely to be spread in a closed room to the controlled release and filtration circuits, and secondly, it creates a slight under pressure in such a closed room in order to avoid the transfer of radioactivity into other areas. This form of containment is called ‘dynamic containment’.

The incident started with the loss of the electrical power supply to the normal ventilation system. The emergency ventilation system, which should have taken over, did not start up. Subsequent investigation indicated that the breakdown of the normal ventilation system and the failure of the emergency ventilation system to come into operation were linked to the presence of a common mode between the electrical power supplies to these ventilation systems. The alarm was signalled in the guard post, but the information reached neither the supervisory staff nor the operating personnel.

The operating personnel were only informed that the alarm had been triggered just over one hour after the shift had started.

The results of measurements of atmospheric contamination taken at all the work stations being monitored did not provide any evidence of an increase in atmospheric contamination.

Rating description

The ventilation system was designed to cascade air flows from areas of low contamination to areas of successively higher or potentially higher contamination. Had there been a coincident event leading to pressurization, some radioactivity which should otherwise have been discharged via a filtration system would be discharged to the plant operating area and then to the atmosphere without the same degree of filtration. The maximum potential consequence would be:

— On site: level 3 (widespread air contamination),
— Off site: level 4.

The maximum defence in depth rating is therefore level 2.

The remaining independent safety provisions, not including ultimate emergency procedures, are:

— Installed (automatic) fire fighting systems,
— The building structure which provided both containment and decontamination to reduce exposures,
— The lack of fuel fire.

Following Section IV–3.2.2.3, there were more than two effective safety layers and a basic rating of 0 is therefore appropriate. However, the OL&C were violated (work continued without ventilation) and thus the event is uprated to level 1.

Example 18: Loss of ventilation in a fission product storage facility — level 1

Event description

The containment of high level liquid waste was provided by:

— The vessels;
— Two separate 100% extract ventilation systems which provide dynamic confinement, avoiding any transfer of radioactivity into areas and directing radioactivity likely to be spread to treatment and filtration circuits;
— Cooling safety systems to avoid boiling;
— Pulsed safety systems to avoid hot points in the vessels owing to deposition of solid particles;
—Specific extract ventilation system assuring the collection of hydrogen to prevent an explosion.

The event occurring was a total shutdown of the extract ventilation systems. The pressure gradient between the cells and other areas was not assured for about three hours. However, the safety provisions to maintain the dilution of hydrogen proceeded normally (pressure air vessel and availability of nitrogen bottles).

Rating description

The ventilation system is required for three purposes:

(a) Maintenance of the hydrogen concentration below the lower explosive limit;
(b) Control of radioactive discharges via a filtered route;
(c) Maintenance of pressure gradients between vessels, cells and plant operating areas.

Prolonged loss of ventilation with a fire or explosion in the vessel ventilation system could give rise to:

— Increased doses to operators, maximum level 2 via pressurization.
— Widespread air contamination, maximum level 3.
— Increased discharges to the atmosphere via cell ventilation routes which have lower levels of filtration than vessel ventilation routes. Maximum consequences may exceed level 4.
— Plant damage, but with radioactive materials fully recoverable and contained (level 4).

The safety layers remaining are:

— Vessel cooling, which limits the evolution rate of gaseous discharges together with H_2 concentration measurement and alarms with the availability of nitrogen to lower the oxygen content if the hydrogen concentration starts to increase.
— The absence of a mechanism to initiate deflagration or detonation.
— Intact cell ventilation filtration systems distant from the vessel and building and cell structures to act as a containment and decontamination system to reduce the impact of discharges.

Following Section IV–3.2.2.3, the maximum potential consequences are level 5 and there are three safety layers available. The basic rating is therefore level 1 and there are no reasons to uprate the level.

Example 19: Lost sealed source — level 2

Event description

A 2 GBq ^{226}Ra source, used for functional testing of instrumentation, was found to be missing from its shielded transport container during the testing of a series of radiation monitors. The source was found within a controlled area, lying in a corridor freely accessible to personnel.

Rating explanation

Such a source would deliver 80 Sv/h at 1 cm, clearly enough to cause burns (level 3) within a few minutes of exposure or a fatality. The maximum rating under defence in depth was therefore level 2. Given the short time, all potential safety layers were rendered ineffective. The rating is therefore level 2.

Example 20: Spillage of plutonium contaminated liquid onto a laboratory floor — level 2

Event description

A flexible hose feeding cooling water to a glass condenser in a glove box became detached. Water flooded the glove box and filled an ambidextrous glove until the glove burst. The spilled water contained about 2.3 GBq of ^{239}Pu.

Rating description

The laboratory was not designed to contain spillages. Liquid spillages are assessed on the radiological equivalence of a few hundred GBq of ^{106}Ru.

From Section III–2.4,

1	Bq ^{239}Pu	≡	3000 Bq ^{106}Ru
2.3	GBq ^{239}Pu	≡	6.9×10^3 GBq ^{106}Ru

The quantity spilled is greater than the level 2 quantity, but less than the level 3 quantity of a few thousand TBq. Because the spillage occurred as a liquid there is little likelihood of any significant exposure of personnel.

Example 21: Supposedly empty shipping containers found to contain nuclear material — level 1

Event description

A fuel manufacturing plant receives from overseas uranium oxide enriched in ^{235}U. The material travels in special cans mechanically sealed within a sea container. After removing the material, the fuel manufacturer sends the empty cans back to the provider.

Upon receiving a container of 150 cans that were supposedly empty, the uranium oxide provider discovered that two cans were full containing in all 100 kg of uranium oxide. The estimated activity of the material was 8^9 Bq; however, the outer surface of the cans and the sea container were found to be clean. No worker or member of the public received any unanticipated dose from this event.

Rating description

Although the packaging for empty cans was the same as if they were full (the mechanical seal remained as well as container conditions), labelling of the transport was less demanding and precautions for handling were slightly relaxed. Therefore, there was a breach of OL&C and (according to Section V–1.11) the event is rated at level 1.

Example 22: Complete loss of shutdown cooling — level 1

Event description

The shutdown cooling of the reactor vessel was completely lost for several hours when the suction isolation valves of the RHR system, which was in operation, automatically closed. These valves closed due to the loss of the power supply to Division 2 of the nuclear safety protection system. The alternate power supply was unavailable because of maintenance. The unit had been in the shutdown condition for a long time (about 16 months) and the decay heat was very low. During the period of time the shutdown cooling was unavailable, water in the reactor vessel began to heat up at a rate of approximately 0.3°C per hour. The RHR system was restarted approximately six hours after the initial event.

Rating explanation

Since the reactor was in the shutdown condition, the event has to be rated using the layers approach.

(a) For this particular event, a very long time was available before any significant consequences such as a core degradation or significant radiological releases could occur. This available time allows implementation of a wide range of measures to correct the situation and can therefore be considered as a 'high integrity layer', as mentioned in Section IV–3.2.2.1. As a result of the presence of this high integrity layer, the basic rating of the event is level 0.
(b) Assuming that the configuration was outside the requirements of the OL&C in respect of time allowed to recover, it would lead to a rating at level 1.
(c) If the decay heat had not been very low, the available time would have been much shorter and it could not have been considered as a high integrity layer. In such a case, the effective safety layers are the following:

— Procedures and operator actions to restore the power supply to Division 2 of the nuclear safety protection system;
— Procedures and operator actions to restore the RHR cooling with alternative systems.

The maximum potential consequences for the considered installation lead to a level 5 and above, so the first column of Table V has to be applied. Since there were two layers remaining, the event would have then been rated at level 2.

Example 23: Power excursion at a research reactor during fuel loading — level 2

Event description

A power excursion, which resulted in a reactor trip on overpower, occurred at a pool type research reactor during a refuelling operation. The reactor is currently operating at 2 MW. Following replacement of a shim safety rod control assembly, the fuel assemblies were being returned to the core. After loading the fifth fuel assembly, the shim safety rods were withdrawn to check that the reactor was not critical. The rods were then driven to the 85% withdrawn position instead of the required 40% (safeguard position). On insertion of the sixth fuel assembly, a blue glow was seen and the reactor tripped on overpower. The Log N trip had been bypassed to avoid spurious trips while moving irradiated fuel into position for loading into the core and the bypass had not been turned off. The power transient maximum was estimated to be about 300% full power. Procedures related to refuelling are being reviewed and revised.

Rating explanation

The introduction to Section 3.2. states that the safety layers approach should be used to assess research reactors. The first step is therefore to identify the

maximum potential consequences. This had been assessed for this reactor and it had been shown that the maximum potential rating for this reactor would not exceed level 4. The one barrier preventing a significant release was the overpower trip. Details of that protection are not provided, but unless it can be shown that there are two or more redundant layers of protection, effective under the prevailing operating conditions, it should be assumed that there was only one layer preventing a significant release. The rating from Table V is therefore level 2.

Part VI

APPENDICES

Appendix I

CALCULATION OF RADIOLOGICAL EQUIVALENCE

I.1. INTRODUCTION

This appendix gives multiplying factors which can be applied to the activity released of a specified radionuclide to give an activity that may be compared with those given for ^{131}I. Values of inhalation coefficients have recently been published and these are incorporated into the IAEA Basic Safety Standards (BSS)[8]. They have been used in this analysis.

I.2. METHOD

Comparable scenarios and methodology were used as in the case of the previous INES guidance. They are summarized below.

(a) For off-site impact, the following two pathways were considered:

- Inhalation dose (effective, adult member of the public) from airborne radionuclide concentration, with a breathing rate of 3.3×10^{-4} $m^3 \cdot s^{-1}$ and an inhalation dose coefficient (D_{inh}, $Sv \cdot Bq^{-1}$);
- External gamma radiation dose (effective, adult), integrated over 50 a, from ground deposited radionuclides. Ground deposition is related to airborne concentration using deposition velocities (Vg) of 10^{-2} $m \cdot s^{-1}$ for elemental iodine and 1.5×10^{-3} $m \cdot s^{-1}$ for other materials. The integrated dose over 50 a from unit ground deposition of each radionuclide is used (D_{gnd}, Sv per $Bq \cdot m^{-2}$) and a factor of 0.5 is applied to this to take account of the roughness of the ground.

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[8] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANISATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, WORLD HEALTH ORGANIZATION, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, IAEA, Vienna (1996).

The total dose (D_{tot}) resulting from an activity release Q and the time integrated, ground level airborne radionuclide concentration of X (Bq·s·m^{-3} per Bq released) is:

$$D_{tot} = QX(D_{inh}\cdot\text{breathing rate} + Vg\ D_{gnd}\ 0.5)$$

For each radionuclide, the relative radiological equivalence to ^{131}I can therefore be calculated as the ratio of the respective values of $D_{tot}/(QX)$.

(b) The on-site consequences consider only the inhalation pathway, and the inhalation coefficients are for workers.

I.3. BASIC DATA

The inhalation coefficients in the second and third columns of Table VIII were taken from the BSS (see footnote 8), apart from U_{nat} which is not listed in that document. Values for U_{nat} were calculated by summing the contributions from ^{238}U, ^{235}U, ^{234}U and their main decay products, as given below. Where a radionuclide has a number of lung absorption types, the maximum value of the inhalation coefficient was used.

The 50 a integrated doses from external gamma radiation were calculated by the National Radiological Protection Board, United Kingdom. The data for ^{235}U include ^{231}Th, and those for ^{238}U include ^{234}Th and ^{234}Pa^{m}. The natural uranium values were calculated using the following ratios: ^{234}U (48.9%), ^{235}U (2.2%) and ^{238}U (48.9%).

I.4. RESULTS

The multiplying factors applicable for on-site impact are obtained by dividing the value for each radionuclide by that for ^{131}I. These are given in Table IX, and in rounded form in Table X. The multiplying factors are within a factor of a few from those published in the previous INES clarification document.[9]

The calculation of the multiplying factors applicable to off-site impact is shown in Table XI. The external dose per Bq·s·m^{-3} (fourth column) is added to the dose from inhalation (seventh column) to give the total for the two pathways (eighth column). The total value for each radionuclide is divided by that for ^{131}I to give the multiplying factors listed in the final column. These are given in rounded form in Table X. The multiplying factors are within a factor of a few from those published in the clarification document.[9]

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[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Clarification of Issues Raised: Addendum to the INES User's Manual, IAEA, Vienna (1996).

TABLE VIII. BASIC DATA

Nuclide	Inhalation coefficients		External from deposit	
	Sv/ Bq (workers) (from footnote 8)	Sv/Bq (public) (from footnote 8)	$Sv \cdot h^{-1}$ per $Bq \cdot m^{-2}$ (a)	$Sv \cdot 50\ a^{-1}$ per $Bq \cdot m^{-2}$ (a)
^{131}I	1.10×10^{-8}	7.40×10^{-9}	—	2.48×10^{-10}
HTO	1.80×10^{-11}	2.60×10^{-10}	—	0
^{32}P	2.90×10^{-9}	3.40×10^{-9}	—	0
^{54}Mn	1.20×10^{-9}	1.50×10^{-9}	—	1.96×10^{-8}
^{60}Co	1.70×10^{-8}	3.10×10^{-8}	—	2.30×10^{-7}
^{99}Mo	1.10×10^{-9}	9.90×10^{-10}	—	5.57×10^{-11}
^{137}Cs	6.70×10^{-9}	3.90×10^{-8}	—	1.25×10^{-7}
^{134}Cs	9.60×10^{-9}	2.00×10^{-8}	—	7.24×10^{-8}
^{132}Te	3.00×10^{-9}	2.00×10^{-9}	—	6.49×10^{-10}
^{90}Sr	7.70×10^{-8}	1.60×10^{-7}	—	0
^{106}Ru	3.50×10^{-8}	6.60×10^{-8}	—	5.27×10^{-9}
$^{234}U(S)$[b]	6.80×10^{-6}	9.40×10^{-6}	3.40×10^{-16}	1.49×10^{-10}
$^{235}U(S)$[b]	6.10×10^{-6}	8.50×10^{-6}	3.65×10^{-13}	1.60×10^{-7}
$^{235}U(M)$[b]	1.80×10^{-6}	3.10×10^{-6}	3.65×10^{-13}	1.60×10^{-7}
$^{235}U(F)$[b]	6.00×10^{-7}	5.20×10^{-7}	3.65×10^{-13}	1.60×10^{-7}
$^{238}U(S)$[b]	5.70×10^{-6}	8.00×10^{-6}	5.36×10^{-14}	2.35×10^{-8}
$^{238}U(M)$[b]	1.60×10^{-6}	2.90×10^{-6}	5.36×10^{-14}	2.35×10^{-8}
$^{238}U(F)$[b]	5.80×10^{-7}	5.00×10^{-7}	5.36×10^{-14}	2.35×10^{-8}
U_{nat}	6.20×10^{-6}	8.70×10^{-6}	3.44×10^{-14}	1.51×10^{-8}
^{239}Pu	1.00×10^{-4}	1.20×10^{-4}	1.75×10^{-16}	7.67×10^{-11}
^{241}Am	2.70×10^{-5}	9.60×10^{-5}	3.65×10^{-14}	1.60×10^{-8}

[a] Calculation of radiological equivalence for the INES User's Manual, letter from S. Hughes to S.J. Mortin, 2000

[b] Lung absorption types: S — slow; M — medium; F — fast. If unsure, use the most conservative value.

TABLE IX. ON-SITE IMPACT, INHALATION ONLY

Nuclide	Inhalation coefficient (Sv/Bq) (workers)	Ratio to ^{131}I
^{131}I	1.10×10^{-8}	1.0
HTO	1.80×10^{-11}	0.002
^{32}P	2.90×10^{-9}	0.3
^{54}Mn	1.20×10^{-9}	0.1
^{60}Co	1.70×10^{-8}	1.5
^{99}Mo	1.10×10^{-9}	0.1
^{137}Cs	6.70×10^{-9}	0.6
^{134}Cs	9.60×10^{-9}	0.9
^{132}Te	3.00×10^{-9}	0.3
^{90}Sr	7.70×10^{-8}	7.0
^{106}Ru	3.50×10^{-8}	3.2
$^{235}U(S)$[a]	6.10×10^{-6}	554.5
$^{235}U(M)$[a]	1.80×10^{-6}	163.6
$^{235}U(F)$[a]	6.00×10^{-7}	54.5
$^{238}U(S)$[a]	5.70×10^{-6}	518.2
$^{238}U(M)$[a]	1.60×10^{-6}	145.5
$^{238}U(F)$	5.80×10^{-7}	52.7
U_{nat}	6.20×10^{-6}	563.6
^{239}Pu	1.00×10^{-4}	9090.9
^{241}Am	2.70×10^{-5}	2454.5

[a] Lung absorption types: S — slow; M — medium; F — fast. If unsure, use the most conservative value.

TABLE X. OFF-SITE IMPACT, INHALATION AND EXTERNAL DOSE FROM GROUND DEPOSIT

Nuclide	External 50 a dose (Sv per Bq·m^{-2})	Deposition velocity, *Vg* (m·s^{-1})	External 50 a dose (Sv per Bq·s·m^{-3})	Inhalation coefficient (public) (Sv per Bq)	Breathing rate (m^3·s^{-1})	Inhalation dose Sv per (Bq·s·m^{-3})	Total dose Sv per (Bq·s·m^{-3})	Ratio to ^{131}I
^{131}I	2.48×10^{-10}	1.00×10^{-2}	1.24×10^{-12}	7.40×10^{-9}	3.30×10^{-4}	2.44×10^{-12}	3.68×10^{-12}	1.0
HTO	0	0	0	2.60×10^{-10}	3.30×10^{-4}	8.58×10^{-14}	8.58×10^{-14}	0.02
^{32}P	0	1.50×10^{-3}	0	3.40×10^{-9}	3.30×10^{-4}	1.12×10^{-12}	1.12×10^{-12}	0.30
^{54}Mn	1.96×10^{-8}	1.50×10^{-3}	1.47×10^{-11}	1.50×10^{-9}	3.30×10^{-4}	4.95×10^{-13}	1.52×10^{-11}	4.1
^{60}Co	2.30×10^{-7}	1.50×10^{-3}	1.73×10^{-10}	3.10×10^{-8}	3.30×10^{-4}	1.02×10^{-11}	1.83×10^{-10}	49.6
^{99}Mo	5.57×10^{-11}	1.50×10^{-3}	4.18×10^{-14}	9.90×10^{-10}	3.30×10^{-4}	3.27×10^{-13}	3.68×10^{-13}	0.1
^{137}Cs	1.25×10^{-7}	1.50×10^{-3}	9.38×10^{-11}	3.90×10^{-8}	3.30×10^{-4}	1.29×10^{-11}	1.07×10^{-10}	29.0
^{134}Cs	7.24×10^{-8}	1.50×10^{-3}	5.43×10^{-11}	2.00×10^{-8}	3.30×10^{-4}	6.60×10^{-12}	6.09×10^{-11}	16.5
^{132}Te	6.49×10^{-10}	1.50×10^{-3}	4.87×10^{-13}	2.00×10^{-9}	3.30×10^{-4}	6.60×10^{-13}	1.15×10^{-12}	0.3
^{90}Sr	0	1.50×10^{-3}	0	1.60×10^{-7}	3.30×10^{-4}	5.28×10^{-11}	5.28×10^{-11}	14.3
^{106}Ru	5.27×10^{-9}	1.50×10^{-3}	3.95×10^{-12}	6.60×10^{-8}	3.30×10^{-4}	2.18×10^{-11}	2.57×10^{-11}	7.0
^{235}U(S)[a]	1.60×10^{-7}	1.50×10^{-3}	1.20×10^{-10}	8.50×10^{-6}	3.30×10^{-4}	2.81×10^{-9}	2.92×10^{-9}	794.4
^{235}U(M)[a]	1.60×10^{-7}	1.50×10^{-3}	1.20×10^{-10}	3.10×10^{-6}	3.30×10^{-4}	1.02×10^{-9}	1.14×10^{-9}	310.4
^{235}U(F)[a]	1.60×10^{-7}	1.50×10^{-3}	1.20×10^{-10}	5.20×10^{-7}	3.30×10^{-4}	1.72×10^{-10}	2.92×10^{-10}	79.2
^{238}U(S)[a]	2.35×10^{-8}	1.50×10^{-3}	1.76×10^{-11}	8.00×10^{-6}	3.30×10^{-4}	2.64×10^{-9}	2.66×10^{-9}	721.8
^{238}U(M)[a]	2.35×10^{-8}	1.50×10^{-3}	1.76×10^{-11}	2.90×10^{-6}	3.30×10^{-4}	9.57×10^{-10}	9.75×10^{-10}	264.7
^{238}U(F)[a]	2.35×10^{-8}	1.50×10^{-3}	1.76×10^{-11}	5.00×10^{-7}	3.30×10^{-4}	1.65×10^{-10}	1.83×10^{-10}	49.6
U_{nat}	1.51×10^{-8}	1.50×10^{-3}	1.13×10^{-11}	8.70×10^{-6}	3.30×10^{-4}	2.87×10^{-9}	2.88×10^{-9}	782.8
^{239}Pu	7.67×10^{-11}	1.50×10^{-3}	5.75×10^{-14}	1.20×10^{-4}	3.30×10^{-4}	3.96×10^{-8}	3.96×10^{-8}	10755.0
^{241}Am	1.60×10^{-8}	1.50×10^{-3}	1.20×10^{-11}	9.60×10^{-5}	3.30×10^{-4}	3.17×10^{-8}	3.17×10^{-8}	8607.3

[a] Lung absorption types: S — slow; M — medium; F — fast. If unsure, use the most conservative value.

TABLE XI. RADIOLOGICAL EQUIVALENCES

Nuclide	Multiplication factors	
	Off-site impact	On-site impact
^{131}I	1(1)	1(1)
HTO	0.02(–)	0.002(–)
^{32}P	0.3(–)	0.3(–)
^{54}Mn	4(–)	0.1(–)
^{60}Co	50(–)	1.5(–)
^{99}Mo	0.1(–)	0.1(–)
^{137}Cs	30(90)	0.6(1)
^{134}Cs	20(–)	0.9(2)
^{132}Te	0.3(–)	0.3(4)
^{90}Sr	10(30)	7(10)
^{106}Ru	7(10)	3(1)
$^{235}U(S)$[a]	800(–)	600(–)
$^{235}U(M)$[a]	300(–)	200(–)
$^{235}U(F)$[a]	100(–)	50(–)
$^{238}U(S)$[a]	700(2500)	500(1000)
$^{238}U(M)$[a]	300(–)	100(–)
$^{238}U(F)$[a]	50(80)	50(35)
U_{nat}	800	600
^{239}Pu	10 000(9000)	9000(10 000)
^{241}Am	9000(9000)	2000(10 000)

[a] Lung absorption types: S — slow; M — medium; F — fast. If unsure, use the most conservative value.

Note: Values in parentheses are those given in footnote 8.

Appendix II

OVERVIEW OF THE PROCEDURE FOR RATING EVENTS FOR REACTORS AT POWER UNDER DEFENCE IN DEPTH

II.1. BACKGROUND

Defence in depth can be considered in a number of different ways. For example, one can consider the number of barriers provided to prevent a release (e.g. fuel, clad, pressure vessel, containment). Equally one can consider the number of systems that would have to fail before an accident could occur (e.g. loss of off-site power plus failure of all essential diesels). It is the latter approach that is adopted within the INES rating procedure.

The basic rating procedure concentrates on the extent of safety system failures, and whether they have been challenged. However, it is recognized that the consequences of all the systems failing can vary considerably. Potential consequences are treated within INES in a relatively simple manner. For events where the maximum potential consequences could be level 5 or higher, level 3 is the maximum appropriate under defence in depth. If the maximum potential consequences of the event cannot be greater than level 4, then the maximum under defence in depth is level 2. Similarly, if the maximum potential consequences cannot exceed level 2, then the maximum under defence in depth is level 1.

We will now consider the approach to rating events in more detail. Two separate but similar approaches are described in the manual. The first, which is summarized here, is most obviously appropriate for events associated with reactors at power. The second is more likely to be appropriate for events related to shutdown reactors, chemical plants, fuel route faults, provisions associated with protection to workers, etc. In general, the approach to be used depends upon the manner in which the safety of the plant has been assessed.

II.2. PROCEDURE FOR EVENTS ASSOCIATED WITH REACTORS AT POWER

Consider a plant where the protection against loss of off-site power is provided by four essential diesels. In order for an accident to occur, the event must challenge plant safety (e.g. LOOP) and the protection must fail (e.g. all diesels fail to start). The initial challenge to plant safety (LOOP in the example) is termed the ‘initiator’ and the response of the diesels is defined by the ‘Operability of the safety function’

(post-trip cooling in this example). Thus, for an accident to occur there needs to be an initiator and inadequate operability of safety functions.

Defence in depth measures how near we are to that accident, i.e. whether the initiator has occurred, how likely it was and the operability of the safety functions. If off-site power had been lost but all diesels started as intended, an accident was unlikely (such an event would probably be rated at level 0). Similarly, if one diesel had failed under a test but the others were available and off-site supplies were available, then an accident was unlikely (again such an event would probably be rated at level 0).

However, if it was discovered that all diesels had been unavailable for a month, then even though off-site power had been available and the diesels were not required to operate, an accident was relatively likely as the chance of losing off-site power was relatively high (such an event would probably be rated at level 3 provided there were no other lines of protection).

The rating procedure therefore considers whether the safety functions were required to work (i.e. had an initiator occurred), the assumed likelihood of the initiator and the operability of the relevant safety functions.

Appendix III

DERIVATION OF THE TABLES FOR RATING EVENTS FOR REACTORS AT POWER (SECTION IV–3.2.1)

III.1. INCIDENTS INVOLVING A DEGRADATION OF SAFETY SYSTEMS WITHOUT AN INITIATOR (SECTION IV–3.2.1.3(b))

The categorization of an incident will depend primarily on the extent to which the safety functions are degraded and on the likelihood of the initiator for which they are provided. Strictly speaking, the latter is the likelihood of the initiator occurring during the period of safety function degradation since the period of inoperability will vary from one incident to another. Accordingly, if the period of inoperability is very short, a level lower than that provided in the table may be appropriate.

If the operability of a required safety function is inadequate (no matter if it is just inadequate or very inadequate), then an accident was only prevented because the initiator did not occur. For such an incident, if the safety function is required for expected initiators (i.e. those expected to occur once or more during the life of the plant), level 3 is appropriate. If the inadequate safety function is only required for possible or unlikely initiators, a lower level is clearly appropriate because the likelihood of an accident is much lower. For this reason, the table shows level 2 for possible initiators and level 1 for unlikely initiators.

The level chosen should clearly be less when the safety function is adequate than when it is inadequate. Thus, if the function is required for expected initiators, and the operability is just adequate, level 2 is appropriate. However, in a number of cases the safety function operability may be considerably greater than just adequate, but not within OL&C. This is because the minimum operability required by OL&C will often still incorporate redundancy and/or diversity against some expected initiators. In such situations, level 1 would be more appropriate. Thus, the table shows a choice of level 1 or 2. The appropriate value should be chosen depending on the remaining redundancy and/or diversity.

If the safety function is required for possible or unlikely initiators, then reduction by one from the level derived above for an inadequate system gives level 1 for possible initiators and level 0 for less likely initiators. However, it is not considered appropriate to categorize at level 0 a reduction in safety system operability below that required by the OL&C. One important part of defence in depth, a redundant safety system, has been defeated. Thus, level 1 is shown in the table for both possible and unlikely initiators.

If the safety function operability is within the OL&C the plant has remained within its safe operating envelope and level 0 is appropriate for all frequencies of initiators. This is also shown in the table.

III.2. INCIDENTS INVOLVING A REAL INITIATOR (SECTION IV–3.2.1.3(a))

Here the categorization will depend primarily on the operability of the safety functions, but for consistency the same table structure as for events without real initiators is used.

Clearly, if the safety function is inadequate, an accident will have occurred and it may be categorized under off-site or on-site impact. However, in terms of defence in depth, level 3 represents the highest category. This total loss of defence in depth is expressed by 3+ in the table.

If the safety function is just adequate, then again level 3 is appropriate, as a further failure would lead to an accident. However, as noted in the previous section, when inoperability is just less than that required by the operational limits and conditions, it may be considerably greater than just adequate, particularly for expected initiators. Therefore, in the table level 2/3 is shown for expected initiators and adequate safety function, the choice depending on the extent to which the operability is greater than just adequate. For unlikely initiators the operability required by the operational limits and conditions is likely to be just adequate and, therefore, in general level 3 would be appropriate for adequate operability. However, there may be particular initiators for which there is redundancy and therefore the table shows level 2/3 for all initiator frequencies.

If there is full safety function operability and an expected initiator occurs, this should clearly be level 0, as shown in the table. However, occurrences of possible or unlikely initiators, even though there may be considerable redundancy in the safety systems, represent a failure of one of the important parts of defence in depth, namely the prevention of initiators. For this reason the table shows level 1 for possible initiators and level 2 for unlikely initiators.

If the operability of safety functions is the minimum required by OL&C, then in some cases, as already noted, for possible and particularly for unlikely initiators, there will be no further redundancy. Therefore, level 2/3 is appropriate, depending on the remaining redundancy. For expected initiators, there will be additional redundancy and therefore a lower categorization is proposed. The table shows level 1/2, where again the value chosen should depend on the additional redundancy within the safety functions. Where the safety function availability is greater than the minimum required by OL&C but less than full, there may be considerable redundancy and diversity available for expected initiators. In such cases, level 0 would be more appropriate.

Appendix IV

EXAMPLES OF INITIATORS

IV.1. PRESSURIZED WATER REACTORS (PWR AND WWER)

IV.1.1. Expected

— Reactor trip;
— Inadvertent chemical shim dilution;
— Loss of main feedwater flow;
— Reactor coolant system depressurization by inadvertent operation of an active component (e.g. a safety or relief valve);
— Inadvertent reactor coolant system depressurization by normal or auxiliary pressurizer spray cooldown;
— Power conversion system leakage that would not prevent a controlled reactor shutdown and cooldown;
— Steam generator tube leakage in excess of plant Technical Specifications, but less than the equivalent of a full tube rupture;
— Reactor coolant system leakage that would not prevent a controlled reactor shutdown and cooldown;
— Loss of off-site AC power, including consideration of voltage and frequency disturbances;
— Operation with a fuel assembly in any misoriented or misplaced position;
— Inadvertent withdrawal of any single control assembly during refuelling;
— Minor fuel handling incident;
— Complete loss or interruption of forced reactor coolant flow, excluding reactor coolant pump locked rotor.

IV.1.2. Possible

— Small LOCA,
— Full rupture of one steam generator tube,
— Dropping of a spent fuel assembly involving only the dropped assembly,
— Leakage from spent fuel pool in excess of normal make-up capability,
— Blowdown of reactor coolant through multiple safety or relief valves.

IV.1.3. Unlikely

— Major LOCA, up to and including the largest justified pipe rupture in the reactor coolant pressure boundary;

— Single control rod ejection;
— Major power conversion system pipe rupture, up to and including the largest justified pipe rupture;
— Dropping of a spent fuel assembly onto other spent fuel assemblies.

IV.2. BOILING WATER REACTORS

IV.2.1. Expected

— Reactor trip;
— Inadvertent withdrawal of a control rod during reactor operation at power;
— Loss of main feedwater flow;
— Failure of reactor pressure control;
— Leakage from main steam system;
— Reactor coolant system leakage that would not prevent a controlled reactor shutdown and cooldown;
— Loss of off-site power AC, including consideration of voltage and frequency disturbances;
— Operation with a fuel assembly in any misoriented or misplaced position;
— Inadvertent withdrawal of any single control rod assembly during refuelling;
— Minor fuel handling incident;
— Loss of forced reactor coolant flow.

IV.2.2. Possible

— Small LOCA,
— Rupture of main steam piping,
— Dropping of spent fuel assembly involving only the dropped assembly,
— Leakage from spent fuel pool in excess of normal make-up capability,
— Blowdown of reactor coolant through multiple safety or relief valves.

IV.2.3. Unlikely

— Major LOCA, up to and including the largest justified pipe rupture in the reactor coolant pressure boundary;
— Single control rod drop;
— Major rupture of main steam pipe;
— Dropping of a spent fuel assembly onto other spent fuel assemblies.

IV.3. CANDU PRESSURIZED HEAVY WATER REACTORS

IV.3.1. Expected

— Reactor trip;
— Inadvertent chemical shim dilution;
— Loss of main feedwater flow;
— Loss of reactor coolant system pressure control (high or low) owing to failure or inadvertent operation of an active component (e.g. feed, bleed or relief valve);
— Steam generator tube leakage in excess of plant operating specification but less than the equivalent of a full tube rupture;
— Reactor coolant system leakage that would not prevent a controlled reactor shutdown and cooldown;
— Power conversion system leakage that would not prevent a controlled reactor shutdown and cooldown;
— Loss of off-site power AC, including consideration of voltage and frequency disturbances;
— Operation with fuel bundle(s) in any misplaced position;
— Minor fuel handling incident;
— Reactor coolant pump(s) trip;
— Loss of main feedwater flow to one or more steam generators;
— Flow blockage in an individual channel (less than 70%);
— Loss of moderator cooling;
— Loss of computer control;
— Unplanned regional increase in reactivity.

IV.3.2. Possible

— Small LOCA (including pressure tube rupture),
— Full rupture of one steam generator tube,
— Blowdown of reactor coolant through multiple safety or relief valves,
— Damage to irradiated fuel or loss of cooling to fuelling machine containing irradiated fuel,
— Leakage from irradiated fuel bay in excess of normal make-up capability,
— Feedwater line break,
— Flow blockage in an individual channel (more than 70%),
— Moderator failure,
— Loss of end shield cooling,
— Shutdown cooling failure,
— Unplanned bulk increase in reactivity,

— Loss of service water (low pressure, high pressure service water or recirculated cooling water),
— Loss of instrument air,
— Loss of on-site electrical power (Class IV, III, II or I).

IV.3.3. Unlikely

— Major LOCA, up to and including the largest justified pipe rupture in the reactor coolant pressure boundary;
— Major power conversion system pipe rupture, up to and including the largest justified pipe rupture.

IV.4. RBMK REACTORS (LWGR)

IV.4.1. Expected

— Reactor trip;
— Malfunction in the system of neutron control of reactor power;
— Loss of main feedwater flow;
— Reactor coolant system (primary circuit) depressurization owing to inadvertent operation of an active component (e.g. a safety or relief valve);
— Primary circuit leak not hindering normal reactor trip and cooldown;
— Reduced coolant flow through a group of fuel channels and reactor protection system channels;
— Reduced helium mixture flow in the reactor graphite stacking;
— Loss of off-site AC power, including voltage and frequency disturbances;
— Operation with a fuel assembly in any misoriented or misplaced position;
— Minor fuel handling incident;
— Depressurization of the fuel channel in the course of refuelling.

IV.4.2. Possible

— Small LOCA,
— Spent fuel assembly drop,
— Leakage from spent fuel pool in excess of normal make-up capability,
— Primary coolant leak through multiple safety or relief valves,
— Fuel channel or RPS channel rupture,
— Loss of water flow in any fuel channel,
— Loss of water flow in RPS cooling circuit,
— Total loss of helium mixture flow in the reactor graphite stacking,

— Emergency in the course of on-load refuelling machine operation,
— Total loss of auxiliary power,
— Unauthorized supply of cold water from emergency core cooling system into reactor.

IV.4.3. Unlikely

— Major LOCA up to and including the largest justified pipe rupture in the reactor coolant pressure boundary;
— Main steam pipe break before the main steam isolation valve, including the largest justified pipe rupture;
— Dropping of a spent fuel assembly onto other spent fuel assemblies;
— Total loss of service water flow;
— Fuel assembly ejection from the fuel channel, including ejection from the fuel channel while in the refuelling machine.

IV.5. GAS COOLED REACTORS

IV.5.1. Expected

— Reactor trip;
— Loss of main feedwater flow;
— Very small depressurization;
— Boiler tube leak;
— Loss of off-site AC power, including consideration of voltage and frequency disturbances;
— Inadvertent withdrawal of one or more control rods;
— Minor fuel handling incident;
— Some loss of interruption of forced reactor coolant flow.

IV.5.2. Possible

— Minor depressurization,
— Inadvertent withdrawal of a group of control rods,
— Full boiler tube rupture,
— Dropped fuel stringer (AGR only),
— Closure of circulator inlet guide vanes (AGR only),
— Gag closure faults (AGR only).

IV.5.3. Unlikely

— Major depressurization,
— Failure of steam pipework,
— Failure of feed pipework.

Appendix V

RATING OF EVENTS INVOLVING VIOLATION OF OL&C

The ‘operational limits and conditions’ describe the minimum operability of safety systems such that operation remains within the safety requirements of the plant. They may also include operation with reduced safety system availability for a limited time. In some countries, ‘Technical Specifications’ include OL&C and, furthermore, in the event that the OL&C are not met, describe the actions to be taken, including times allowed for recovery and the appropriate fallback state.

If the system availability is within the OL&C but the utility stays more than the allowed time (as defined in the Technical Specification) in that availability state, the event should be rated at level 1 because of deficiencies in safety culture.

If the system availability is discovered to be less than that allowed by the OL&C, even for a limited time, but the operator goes to a safe state in accordance with the Technical Specifications, the event should be rated as described in Section III–3.2, but should not be uprated due to violation of the Technical Specifications. Account should also be taken of the time for which the safety function availability is less than that defined by the OL&C.

In addition to the formal OL&C, some countries introduce into their Technical Specifications further requirements such as limits that relate to the long term safety of components. For events where such limits are exceeded for a short time, level 0 may be more appropriate.

For reactors in the shutdown state, Technical Specifications will again specify minimum availability requirements, but will not generally specify recovery times and fall back states as it is not possible to identify a safer state. The requirement will be to restore the original plant state as soon as possible. In general, plant failures that reduce availability during shutdown should be rated using the safety layers approach and the reduction in plant availability below that required by the Technical Specifications should not be regarded as a violation of OL&C.

This manual was prepared on the basis of experience gained in applying the 1992 edition and the clarification of issues raised. This updating was carried out under the auspices of the INES Advisory Committee, chaired by S. Mortin, Magnox Generation Business Group, British Nuclear Fuels, United Kingdom.

Appendix VI

LIST OF PARTICIPATING COUNTRIES AND ORGANIZATIONS

Argentina
Armenia
Australia
Austria
Bangladesh
Belarus
Belgium
Brazil
Bulgaria
Canada
Chile
China
Costa Rica
Croatia
Czech Republic
Democratic Republic of the Congo
Denmark
Egypt
Finland
France
Germany
Greece
Guatemala
Hungary
Iceland
India
Iran, Islamic Republic of
Ireland
Italy
Japan
Kazakhstan
Korea, Republic of
Kuwait
Lebanon
Lithuania
Luxembourg
Mexico
Netherlands
Norway
Pakistan
Peru
Poland
Portugal
Romania
Russian Federation
Saudi Arabia
Slovakia
Slovenia
South Africa
Spain
Sri Lanka
Sweden
Switzerland
Syrian Arab Republic
Turkey
United Kingdom of Great Britain and Northern Ireland
Ukraine
United States of America
Viet Nam
Yugoslavia, Federal Republic of

INTERNATIONAL LIAISON
European Commission
Nuclear Energy Institute
World Association of Nuclear Operators